INNOVATIVE CONFINEMENT CONCEPTS

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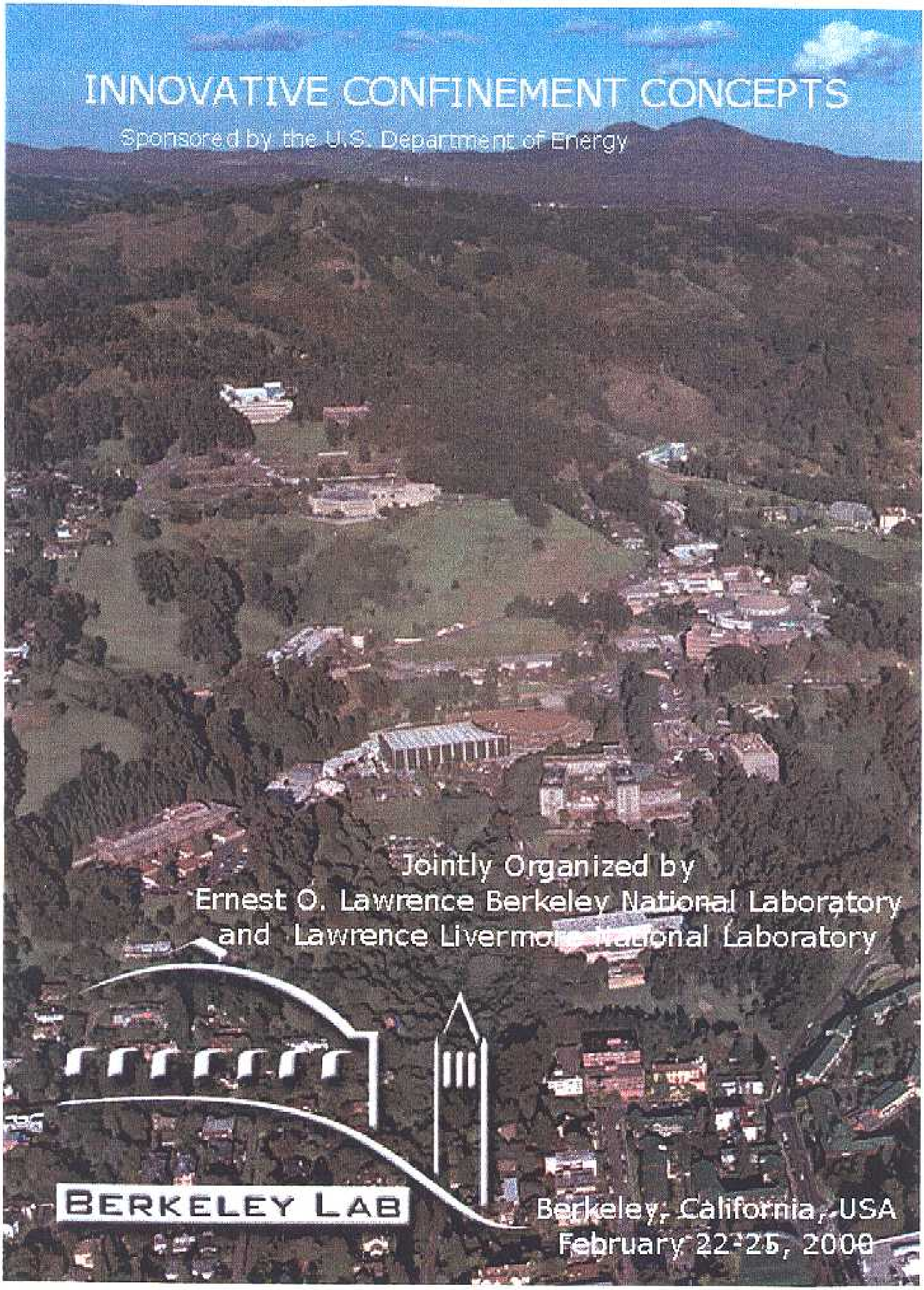

Jointly Organized by
Ernest O. Lawrence Berkeley National Laboratory
and Lawrence Livermore National Laboratory

BERKELEY LAB

Berkeley, California, USA
February 22-25, 2000

LBNL-45208

Innovative Confinement Concepts Workshop

February 22 – 25, 2000

Berkeley, California, USA

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ICC 2000

Welcome to the annual Innovative Confinement Concepts Workshop. The purpose of the workshop is to provide a forum for exchange of ideas and presentation of results concerning innovative concepts for the production of commercial electrical power using nuclear fusion. The program for the workshop can be found on the following pages.

Program Committee:

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Innovative Confinement Concepts Workshop 2000

Tuesday – Building 66 Auditorium

Oral Presentations at the Building 66 Auditorium

8:30 – 8:05 Welcoming remarks

8:35 – 10:15 Stellarators
Chair: G.H. Neilson

8:35 – 8:50	TuA1	C. Alejaldre	"Status of TJ – II Flexible Heliac"
8:55 – 9:10	TuA2	J.N. Talmadge	"Measurements of Magnetic Surfaces and Particle Orbits in HSX"
9:15 – 9:30	TuA3	M.C. Zarnstorff	"NCSX Goals, Design Status, and Projected Plasmas"
9:35 – 9:50	TuA4	J.F. Lyon	"Recent Progress in Development of Low-Aspect-Ratio Quasi-Omnigeneous Stellarators"
9:55 – 10:15	Panel discussion		

10:15 – 10:25 Coffee break

10:25 – 11:25 Advanced Boundary Concepts
Chair: W.M. Nevins

10:25 – 10:40	TuB1	C.L. Olson	"Rep-Rated Z-Pinch Fusion Power Plant Concept"
10:45 – 11:00	TuB2	R. Majeski	"Plans for Liquid Lithium Wall Experiments in CDX-U"
11:05 – 11:25	Panel discussion		

11:25 – 1:00 Lunch

1:00 – 2:20 Field Reversed Configurations
Chair: A.L. Hoffmann

1:00 – 1:30	TuC1	K.E. Miller	"Calculations & Measurements of RMF Current Drive in FRCs"
1:35 – 1:55	TuC2	E.V. Belova	"Numerical Study of Tilt Mode Stability in Field-Reversed Configurations"
2:00 – 2:20	Panel discussion		

2:20 – 2:35 Soda break

2:35 – 4:05 Magnetized Target Fusion
Chair: R.C. Kirkpatrick

2:35 – 3:00	TuD1	R.E. Siemon	"Recent Liner Experiments Confirm Plans For Magnetized Target Fusion"
3:05 – 3:20	TuD2	J.T. Slough	"Pulsed High Density Fusion"
3:25 – 3:40	TuD3	D.D. Ryutov	"A Concept Of A Local Spherical Blanket Applied For MTF Systems"
3:45 – 4:05	Panel discussion		

4:05 – 5:25 Inertial Electrostatic Confinement
Chair: J.S. Sarff

4:05 – 4:20	TuE1	K.R. Umstadter	"Penning Fusion Experiment – Ions (PFX-I)"
4:25 – 4:40	TuE2	S.F. Paul	"Confinement Of High Density Pure Ion Plasma In A Cylindrical Trap"
4:45 – 5:00	TuE3	R. Nebel	"Stability of Virtual Cathodes for the Periodically Oscillating Plasma Sphere (POPS)
5:05 – 5:25	Panel discussion		

8:00 Town Meeting at Berkeley Marina Radisson Hotel (Facilitator: R. Goldston)
"Community Discussion: Peer Review in the ICC Program in 2001"

Innovative Confinement Concepts Workshop 2000

Wednesday

Oral Presentations at the Building 50 Auditorium
Posters at the Cafeteria (Building 54)

8:30 – 10:10 Inertial Fusion Energy
Chair: R.O. Bangerter

8:30 – 8:55	WeA1	J.A. Koch	"Fast-Ignition Inertial Confinement Fusion Research"*
9:00 – 9:15	WeA2	N.J. Fisch	"Compression of High Power Laser Pulses in Plasma"
9:20 – 9:45	WeA3	D.A. Callahan-Miller	"Advances in IFE Targets for Heavy Ion Fusion"*
9:50 – 10:10	Panel discussion		

10:10 – 10:20 Coffee break

10:20 – 11:20 Z Pinches
Chair: C.L. Olson

10:20 – 10:35	WeB1	U. Shumlak	"The Flow-Stabilized Z-Pinch Experiment: ZaP"
10:40 – 10:55	WeB2	H.U. Rahman	"Staged Z-Pinch for Controlled Fusion"
11:00 – 11:20	Panel discussion		

11:20 – 12:45 Lunch

12:45 – 2:15 Spheromaks
Chair: E.B. Hooper

12:45 – 1:05	WeC1	S. Woodruff	"Practical Applications Of Helicity to ICCs"
1:10 – 1:30	WeC2	D.N. Hill	"Overview of Spheromak Formation Experiments on SSPX"
1:35 – 1:50	WeC3	P.M. Bellan	"Formation and self-similar expansion of a spheromak configuration in the absence of a flux-conserving boundary"
1:55 – 2:15	Panel discussion		

2:15 – 3:15 Feedback Stabilization
Chair: R.J. Taylor

2:15 – 2:30	WeD1	G.A. Navratil	"Active Feedback Control of the Resistive Wall Mode in HBT-EP"
2:35 – 2:50	WeD2	R. Fitzpatrick	"Nonlinear Dynamics Of Feedback Modulated Magnetic Islands"
2:55 – 3:15	Panel discussion		

3:15 – 3:40 Soda break + put up posters
3:40 – 5:40 Posters (at the Cafeteria, Building 54)

7:30 Banquet at Le Cheval

Innovative Confinement Concepts Workshop 2000

Thursday

Oral Presentations at the Building 50 Auditorium
Posters at the Cafeteria (Building 54)

8:30 – 9:50	Reverse Field Pinches		
Chair: R. Nebel			
8:30 – 8:45	ThA1	M.J. Schaffer	"Helical RFPS: Theory and Experiment"
8:50 – 9:05	ThA2	D.J.G. Craig	"Generation and Evolution of Plasma Flow in the MST Reversed Field Pinch"
9:10 – 9:25	ThA3	J.S. Sarff	"Highlights of Improved Confinement and Future Plans for MST"
9:30 – 9:50	Panel discussion		
9:50 – 10:00	Coffee Break		
10:00 – 11:20	Tokamak Innovations		
Chair: R. Goldston			
10:00 – 10:15	ThB1	M.T. Kotschenreuther	"Reactor Relevant Belt Pinches"
10:20 – 10:35	ThB2	R.J. Taylor	"Initial Results on ET"
10:40 – 10 55	ThB3	L.E. Zakharov	"The Concept of Tokamaks with the Lithium Walls"
11:00 – 11:20	Panel discussion		
11:20 – 12:45	Lunch		
12:45 – 1:55	Dipoles/Mirrors		
Chair: J. Kesner			
12:45 – 1:10	ThC1	D.T. Garnier	"Status of the LDX Project"
1:15 – 1:30	ThC2	R.F. Ellis	"Centrifugal Confinement for Fusion: The Maryland Centrifugal Torus (MCT)"
1:35 – 1:55	Panel discussion		
1:55 – 3:35	Spherical Tori		
Chair: M. Peng			
1:55 – 2:10	ThD1	T.A. Thorson	"Plasma Formation Studies and Plans for the Pegasus Toroidal Experiment"
2:15 – 2:30	ThD2	M. Ono	"Overview of Experimental Results on NSTX"
2:35 – 2:50	ThD3	R. Raman	"Coaxial Helicity Injection For The Generation Of Non-Inductive Current In NSTX"
2:55 – 3:10	ThD4	S.C. Jardin	"TSC Modeling of NSTX"
3:15 – 3:35	Panel discussion		
3:45 - 5:45	Posters (at the Cafeteria, Building 54) & Soda break		

Innovative Confinement Concepts
Workshop
2000

Friday

Facility Tours

9:00 – 11:00	LBNL Tours
9:00 – 10:00	Tour of LBNL Heavy Ion Fusion Experiments
10:00 – 11:00	Tour of LBNL Advanced Light Source
11:00 – 12:00	Transportation to LLNL
12:00 – 12:30	LLNL Badge Office
12:30 – 1:45	Lunch at LLNL
1:45 – 5:00	LLNL Tours
1:45 – 2:30	Tour of SSPX.
2:30 – 3:15	Tour of Falcon laser and Linac
3:15 - 3:45	Tour of Mecury Laser facility
3:45 – 4:00	NIF safety orientation.
4:00 – 5:00	NIF Tour
5:00 – 6:00	Transportation back to hotel in Berkeley

ICC 2000 Posters

***Please note that the Program Committee would like to have all the posters displayed at both sessions—Wednesday afternoon and Thursday afternoon's sessions. This list shows the session at which the author should be at his/her poster presenting the poster.
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Wednesday, Feb. 23 3:40 – 5:40 Building 54 (Cafeteria)

Stellarators

WeP1 A.K. Sen (Columbia University, U.S.A.), "'Stellarator-Pinch': A Composite Fusion Concept"
WeP2 G.H. Neilson (Princeton Plasma Physics Laboratory, U.S.A.), "NCSX Facility Design Progress"

Reversed Field Pinches

WeP3 B.E. Chapman (U. of Wisconsin at Madison, U.S.A.), "Reduced edge instability and improved confinement in the MST RFP"
WeP4 P.A. Gourdain (UCLA, U.S.A.), "Pinch discharges in the Electric Tokamak"

Dipoles

WeP5 J. Kesner (MIT, U.S.A.), D.T. Garnier, M.E. Mauel (Columbia University, U.S.A.), "Physics Issues for a Plasma Confined in a Dipole Field"
WeP6 Leonid E. Zakharov (Princeton University, PPPL, U.S.A.), "Superconducting Morozov's Ring For Refuelling And Controlling Plasma Profiles"
WeP7 Leonid E. Zakharov (Princeton University, PPPL, U.S.A.), "Warm Morozov's Rings For The Helium Exhaust From The Fusion Reactors"

Field Reversed Configurations

WeP8 S.A. Cohen (Princeton Plasma Physics Lab, U.S.A), A.H. Glasser (Los Alamos National Lab, U.S.A.) and R.D.Milroy (U. of Wash., U.S.A.), "Single-particle heating by rotating magnetic fields in FRCs having closed flux surfaces"
WeP9 H.Y. Guo, A.L. Hoffman, J.T. Slough, R. Brooks, E.A. Crawford, P. Euripides (U. of Washington, U.S.A.), "Translation, Confinement and Current Drive of Field Reversed Configuration in TCS"
WeP10 Richard D. Milroy (U. of Washington, U.S.A.), "An MHD Model of Rotating Magnetic Field Current Drive in an FRC"
WeP11 T. Intrator, M. Tuszewski, R. Kirtpatrick, R. Siemon, D.C. Barnes, J. Hwang (Los Alamos National Laboratory), "Modeling High Density Field Reversed Configurations"
WeP12 H. Ji, M. Yamada, E. Belova, R. Kulsrud, S. Jardin, D. Mikkelsen, S. Zweben (Princeton Plasma Physics Laboratory, U.S.A.) and A. Hassam (U. of Maryland, U.S.A.), "Study of Physics of Compact Toroids in the Proposed SPIRIT Program"
WeP13 J.M. Taccetti, T.P.Intrator, R.Kirtpatrick, , I.Lindemuth, R.Moses, , K.Schoenberg, R. Siemon, , P.J.Turchi, M. Tuszewski, G.Wurden, F.Wysocki (Los Alamos National Laboratory, U.S.A.), and T.Cavazos, S.K.Coffey, J.H.Degnan, M.Frese, D.Gale, T.W.Hussey, G.F.Kiuttu, F.M.Lehr, R.E.Peterkin, N.F.Roderick, E.L.Ruden W.Sommars, R. White (Air Force Research Laboratory, U.S.A.) and B. Pearson (U. Of New Mexico, U.S.A.), "Design and Fabrication of a High Density Field Reversed Configuration"
WeP14 J. Greenly, W.J. Podulka, A.V. Gretchikha (Cornell University, USA), Azimuthal Dynamics of Strong Ion Rings

Magnetized Target Fusion

WeP15 Peter T. Sheehey, Los Alamos National Laboratory, Rickey J. Faehl, Los Alamos National Laboratory, Ronald C. Kirkpatrick, Los Alamos National Laboratory, Irvin R. Lindemuth (Los Alamos National Laboratory, U.S.A.), "Detailed Modeling Of Proposed Liner-On-Plasma Fusion Experiments"

WeP16 D.D. Ryutov (Lawrence Livermore National Laboratory, U.S.A.), "Scaling Laws For Formation And Adiabatic Compression Of Of A High-Density FRC In The MTF Setting"

Inertial Fusion Energy

WeP17 L. Ahle, T. C. Sangster, J. Barnard, G. Craig, A. Friedman, D. P. Grote, E. Halaxa, R. L. Hanks, M. Hernandez, H. C. Kirbie, B. G. Logan, S. M. Lund, G. Mant, A. W. Molvik, W. M. Sharp, C. Williams (Lawrence Livermore National Laboratory, U.S.A.), A. Debeling, W. Fritz (Bechtel Nevada Corporation, U.S.A.), and Craig Burkhart (First Point Scientific, U.S.A.), "The Recirculator Project at LLNL*"

WeP18 J.J. Barnard, L. E. Ahle, R. O. Bangerter, F.M. Bienosek, C. M. Celata, A. Faltens, A. Friedman, D.P. Grote, E. Henestroza, W. B. Herrmannsfeldt, M. de Hoon, J. W. Kwan, E.P. Lee, B.G. Logan, S.M. Lund, W. Meier, A. W. Molvik, T. C. Sangster, P. A. Seidl, W. M. Sharp, (LBNL/LLNL VNL, U.S.A.), "Planning for a Heavy Ion IRE"

WeP19 Edward Lee (Lawrence Berkeley National Laboratory, U.S.A.), "A Mini-Final Focus system for Inertial Fusion Energy"

WeP20 C.M. Celata, F.M. Bieniosek, And A. Faltens, (Lawrence Berkeley National Laboratory, U.S.A), "Beam Splitting in a Heavy Ion IRE and Driver"

WeP21 S. A. Payne, C. Bibeau, R. J. Beach, A. Bayramian, J. C. Chanteloup, C. A. Ebbers, M. A. Emanuel, H. Nakano, C. D. Orth, H. T. Powell, J. E. Rothenberg, K. I. Schaffers, J. A. Skidmore, S. B. Sutton, And L. E. Zapata (Lawrence Livermore National Laboratory, U.S.A.), "Diode-Pumped Solid-State Lasers for Inertial Fusion Energy"

WeP22 J. W. Kwan, E. Henestroza , L. Ahle, D. P. Grote (LBNL/LLNL VNL, U.S.A.), "A New Injector Concept for HIF Induction Linacs"

WeP23 J.F. Latkowski (Lawrence Livermore National Lab., U.S.A.), "Advanced IFE Power Plants Utilizing Fast-Ignited, Tritium-Lean Targets"

WeP24 Arthur W. Molvik, Ralph W. Moir (Lawrence Livermore National Lab., U.S.A.), and Caron Jantzen (U. of California at Berkeley and Lawrence Berkeley National Laboratory, U.S.A.) and Per F. Peterson (U. of California at Berkeley, U.S.A.), "Higher Vacuum Operation Of A Liquid-Walled IFE Chamber"

WeP25 C. Niemann, D. Ponce, S. Yu, (Lawrence Berkeley National Laboratory, USA), "Plasma-Channel-Based Reactor and Final Transport"

Thursday, Feb. 24 3:45 – 5:45 Building 54 (Cafeteria)

Please note that the Program Committee would like to have all the posters displayed at both sessions—Wednesday afternoon and Thursday afternoon's sessions. This list shows the session at which the author should be at his/her poster presenting the poster.

Spherical Tori

ThP1 Martin Peng (Oak Ridge National Lab, on assignment at PPPL, U.S.A.), "Physics Innovations in Spherical Torus Plasmas"

ThP2 P. C. Efthimion, J. C. Hosea, R. Kaita, R. Majeski, C. K. Phillips, G. Taylor, J. R. Wilson, B. Jones, J. Menard, T. Munsat (Princeton Plasma Physics Lab., U.S.A.), "Electron Bernstein Waves (Ebw) In Overdense Plasmas as a New Tool For Advanced Tokamak and Spherical Torus Devices"

ThP3 Craig H. Williams, Stanley K. Borowski, Leonard A. Dudzinski, Albert J. Juhasz (NASA Glenn Research Center, U.S.A.), "A Spherical Torus Nuclear Fusion Reactor Space Propulsion Vehicle Concept for Fast Interplanetary Piloted and Robotic Missions"

Spheromaks

ThP4 C.T. Holcomb, T.R. Jarboea, A.T. Matticka, H. McLean, D.N. Hill (Lawrence Livermore National Laboratory, U.S.A.), "Internal Field Measurements and Magnetic Reconstruction in SSPX"

ThP5 H.S. McLean, A. Ahmed, D. Buchenauer (1), D.N. Hill, E.B. Hooper, B. Stallard, R. D. Wood, S. Woodruff, G. Wurden (2), Z. Wang (2), and the SSPX Team (Lawrence Livermore National Laboratory, U.S.A.), "Particle control experiments in SSPX"

ThP6 B.W. Stallard, S. Woodruff, A. Ahmed, D. Buchenauer, D.N. Hill, C.T. Holcomb, E.B. Hooper, H.S. McLean, R.D. Wood (Lawrence Livermore National Lab., U.S.A.), "Modeling of Spheromak Plasma Buildup in SSPX by Power Balance and Helicity Injection"

Thp7 B.I. Cohen, L.L. LoDestro, L.D. Pearlstein, N. Mattor, and R.H. Bulmer (Lawrence Livermore National Laboratory, U.S.A.), and C.R. Sovinec (Los Alamos National Laboratory, U.S.A.), "Simulations and Modeling of SSPX Plasma Evolution"

ThP8 T. Jarboe (U. of Washington, U.S.A.), "Steady Inductive Helicity Injection Current Drive"

ThP9 Edward C. Morse And Charles W. Hartman (U. Of California at Berkeley, U.S.A.), "Experimental Studies Of Spheromaks At The Berkeley Compact Toroid Experiment"

Tokamak Innovations

Thp10 J.-L. Gauvreau, P.-A. Gourdain, M.W. Kissick, L.W. Schmitz And R.J. Taylor (UCLA, U.S.A.), "ET Magnetic Configuration and Equilibrium"

ThP11 M.W. Kissick, J.-N. Leboeuf, S.C. Cowley, J.M. Dawson (Ucla, U.S.A.), "Simulated Poloidal Rotation Effects On Kink Modes For The Electric Tokamak"

Z Pinches

ThP12 F. Winterberg (U. Of Nevada, U.S.A.), "Laser Ignition Of An Isentropically Compressed Dense Z-Pinch"

ThP13 John S. De Groot (U. of California at Davis, U.S.A.), Rick B. Spielman, Craig L. Olson (Sandia National Laboratories, U.S.A.), and Per F. Peterson (U. Of California at Berkeley; U.S.A.), "Direct Conversion in a Z-Pinch IFE Reactor"

Advanced Boundary Concepts

ThP14 H.L. Rappaport, M. Kotschenreuther, R. Fitzpatrick (U. Of Texas At Austin, U.S.A.), "Motion And Stability Of Liquid Metal Walls In Fusion Reactors"

ThP15 Charles D. Orth (Lawrence Livermore National Laboratory, U.S.A.), "Hydro*Star: A New IFE Concept Using the Fusion Chamber as a Steam Boiler"

Inertial Electrostatic Confinement

ThP16 S.R. Bolger, (USA), "Inertial Confinement Fusion in the Neutral Plasma Imploder"

ThP17 G.H. Miley, J. Nadler (University of Illinois at Urbana-Champaign, USA), "Recent Studies of Star Mode IEC Devices and Possible New Directions"

Other

ThP18 T. Ditmire, J. Zweiback, T. E. Cowan, L. J. Perkins, T. D. de la Rubia, G. Hays, J. Hartley, and H. T. Powell (Lawrence Livermore National Laboratory, U.S.A.) and R.A. Smith (Imperial College of Science, Technology and Medicine, London, UK), "Ultrafast fusion neutron sources produced from laser driven explosions of molecular clusters"

ThP19 V.A.Rantsev-Kartinov, A.B.Kukushkin (Inf Rrc "Kurchatov Institute", Russia), "Wild Cables In Fusion Plasmas (Experiment)"

ThP20 A.B.Kukushkin, V.A.Rantsev-Kartinov (Inf Rrc "Kurchatov Institute", Russia), "Wild Cables In Fusion Plasmas (Theoretical View)"

ThP21 R.F. Post (Lawrence Livermore National Lab., U.S.A.), "Some Applications of the Kinetic Tandem Concept"

ThP22 R. Majeski, S. Bernabei, J. Hosea, J. Menard, C. K. Phillips, J. R. Wilson (Princeton Plasma Physics Laboratory, U.S.A.) And D. B. Batchelor, T. Bigelow, M. D. Carter, D. Rasmussen (Oak Ridge National Laboratory, U.S.A.) And The Nstx Hhfw Group, "Fast Wave Heating For Innovative Concepts"

Abstracts

Status of TJ-II Flexible Heliac

C. Alejaldre on behalf of the TJ-II team

TJ-II is a four period low magnetic shear stellarator, designed with a high degree of experimental flexibility, which is operating in Madrid since 1998 (R = 1.5 m, a < 0.22 m, B0 = 1.2 T, PECRH = 700 kW, PNBI = 3MW under installation)1.

Plasma is until now being created and heated using two ECRH transmission lines with different power densities (1 vs. 25 W/cm2) and steering launching capabilities (fix vs. poloidal and toroidal variation); stationary plasmas during the whole gyrotron pulse (= 0.3 s) with stored energies up to 1.3 kJ and 1.5 keV central electron temperature are routinely achieved.

Using TJ-II flexibility and a set of state of the art diagnostics, a configuration scan has been initiated which shows a significant modification in plasma profiles and stored energies in the device. Stored energy scales with iota and plasma volume.

A particular and important issue for steady-state operation of so called "advance modes" in Tokamaks and to improve/optimize Stellarator performance is the understanding (and control) of transport barrier formation in these systems. In TJ-II, evidence of sheared ExB flows associated with the presence of rational surfaces has been observed in the edge region of the stellarator together with an improvement of particle confinement of high energy electrons. This mechanism may be associated with the widely reported spontaneous formation of transport barriers at rational surfaces in magnetically confined plasmas.

(1) C. Alejaldre et al., Plasma Physics and Controlled Fusion, in press

Presenting Author: C. Alejaldre

Measurements of Magnetic Surfaces and Particle Orbits in HSX

J.N. Talmadge, A. Almagri, D.T. Anderson, F.S.B. Anderson, L. Feldner, S. Gerhardt, J. Radder, V. Sakaguchi, J. Shafii
HSX Plasma Laboratory, University of Wisconsin-Madison

The Helically Symmetric Experiment (HSX), which began initial operation in August 1999, is the world's first stellarator in which a direction of symmetry in the magnetic field is restored to an inherently three-dimensional device. It achieves this symmetry by reducing the toroidal curvature to negligible values, although physically the device has an aspect ratio of 8. The vacuum rotational transform in HSX is just above one; however, the 'effective transform' is about three times larger. This effective transform is given by N-m*iota, where N, the magnetic field toroidal mode number is 4 and m, the poloidal mode number is 1. The symmetry in the magnetic field and the high effective transform are responsible for a number of unique properties for a stellarator including: small passing particle drifts and banana widths, greatly reduced direct loss orbits, small Pfirsch-Schluter and bootstrap currents, high equilibrium beta limit and low parallel viscous damping in the direction of symmetry.

Vacuum magnetic surfaces were measured in HSX using a low-energy (< 100 eV) electron beam in a steady-state magnetic field of 1 kG. A fluorescent copper-wire mesh intercepts the beam and a CCD camera captures the resulting image. The digitized image is then corrected for geometrical distortion and the data is compared to numerical calculations of magnetic surfaces. The experimental results show closed, nested surfaces inside the separatrix with no evidence of magnetic islands due to error fields. The measured rotational transform agrees with the numerical calculations to within 1%.

In a quasihelical field, such as in HSX, the deviation of a passing particle from a flux surface should be reduced by the factor N-m*iota ~ 3 compared to a particle in a device with only toroidal curvature. Measurements of the scaling of the particle shift with magnetic field and particle energy, as well as the direction of the orbit shift with respect to the flux surface, can potentially verify whether or not the toroidal curvature in HSX is indeed very small. At lower magnetic fields and higher beam energies, the same detection system used to map the vacuum magnetic surfaces can also record the drift orbits of passing electrons. Preliminary measurements in HSX are in agreement with a reduced toroidal curvature and a small deviation of the particle drift from a flux surface.

This research is supported by the US DOE under grant DE-FG02-93ER54222.

Presenting Author: J.N. Talmadge

TuA3

NCSX Goals, Design Status, and Projected Plasmas

M.C. Zarnstorff, A. Brooks, G.-Y. Fu, R. Goldston, L.-P. Ku, Z. Lin, R. Majeski, D. Monticello. H. Mynick, N. Pomphrey, M. Redi, W. Reiersen, J. Schmidt
Princeton Plasma Physics Lab.
S. Hirshman, W. Houlberg, J. Lyon, D. Spong
Oak Ridge National Lab.
A. Boozer
Columbia University
W. Miner, P. Valanju, University of Texas at Austin

Candidate configurations for the NCSX quasi-axisymmetric have been designed to achieve beta~4% at aspect ratio A~3.4 with passive stability to the ballooning, external kink, vertical, and neoclassical-tearing instabilities via 3D shaping, without needing a conducting wall or active feedback. The proposed experiment will test these theoretical stability predictions and reliable operation at the beta-limit without disruptions. The quasi-axisymmetric optimization of the magnetic field gives orbit-confinement and neoclassical transport similar to tokamaks and should allow tokamak-like manipulation of the ExB flow-shear. Novel simple coil designs have been developed and coil-set flexibility is being explored and designed. Target plasma parameters have been projected using empirical scaling and numerical calculations of fast ion and thermal plasma neoclassical losses. They indicate that the beta~4% goal can be obtained, assuming a confinement enhancement of ~2.3 times ISS-95 scaling or 1.6 times ITER-89P scaling.

Presenting Author: M.C. Zarnstorff

TuA4

Recent Progress in Development of Low-Aspect-Ratio Quasi-Omnigeneous Stellarators

J.F. Lyon, L.A. Berry, S.P. Hirshman, D.A. Spong, R. Sanchez,
D.J. Strickler, A.S. Ware, J.C. Whitson, D.B. Batchelor, B.E. Nelson
Oak Ridge National Laboratory

In quasi-omnigeneous (QO) stellarators, bounce-averaged particle drift surfaces are approximately aligned with flux surfaces to reduce energetic orbit losses and neoclassical energy transport. This approach has the potential for a low-aspect-ratio ($A_p = R_0/a_p$) stellarator with good confinement, high beta, and a small bootstrap current (typically ~1/10 that in a comparable tokamak), which leads to configurations that are relatively insensitive to beta and should be robust against current-driven modes (external kinks), vertical instabilities, and disruptions. These stellarators have some general similarity to the drift-optimized W 7-X "helias" configuration, but the QO configurations studied here have: (1) a factor of 3-4 smaller plasma aspect ratio; (2) non-zero bootstrap current; (3) a factor of three larger helical field component, which allows higher vacuum rotational transform $\iota\!\!\!-$ at low A_p without relying on a plasma current; and (4) a mirror component of the magnetic field that varies strongly with radius, which produces a poloidal ∇B drift that reduces transport and leads to closed drift surfaces for trapped particles, even at low beta.

QO configurations with field periods N_{fp} = 3 and 4 and A_p from 3 to 4.8 are optimized with respect to trapped particle confinement (poloidal variation of B_{max}, B_{min}, J*, J, and local transport), β limits (global magnetic well, Mercier stability, ballooning stability), amount of bootstrap current, magnetic field ripple, outer surface curvature. The development in the last year of improved QO configurations with a stronger physics basis is due to incorporation of fast calculations of the bootstrap current and the ballooning stability limit within the iterative optimization loop. Typical QO parameters are $A_p = 3.6$, $\iota\!\!\!-(0) = 0.55$, and $\iota\!\!\!-(a_p) = 0.64$ for $N_{fp} = 3$ and $A_p = 4.2$, $\iota\!\!\!-(0) = 0.68$, and $\iota\!\!\!-(a_p) = 0.78$ for $N_{fp} = 4$. The calculated energy confinement time shows a strong dependence on electric field with $\tau_E^{neo}/\tau_E^{ISS95}$ = 3-5 for typical values of E_r. Self-consistent bootstrap current and good transport are obtained for a low-A_p N_{fp} = 3 configuration with a ballooning stability β limit >3%. Current studies are focusing on obtaining more reactor-relevant configurations with a higher β limit. Modular coils with adequate distances between the plasma and the coils and between adjacent coils for plasma heating and diagnostic access have been found that recreate the optimized magnetic configuration. Engineering considerations for these configurations are being examined.

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Presenting Author: J.F. Lyon

Rep-Rated Z-Pinch Fusion Power Plant Concept

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The Z machine at SNL is the most powerful multi-module synchronized pulsed-power accelerator in the world. Rapid development of z-pinch loads on Z has led to outstanding progress in the last few years, resulting in radiative powers of up to 280 TW in 4 ns and a total radiated x-ray energy of 1.8 MJ. Presently, demonstration of a single-shot, high-yield fusion target is a goal of the z-pinch ICF program at SNL. For Inertial Fusion Energy (IFE), a rep-rated fusion capability is needed. Recent developments have led to a viable conceptual approach for a rep-rated z-pinch power plant for IFE, which is the subject of this presentation.

The rep-rated z-pinch concept exploits the advantages of going to high yield (~ few GJ) at low rep-rate (~ 0.1 Hz), and using a Recyclable Transmission Line (RTL) to provide the necessary standoff between the fusion target and the power plant chamber. The RTL would be cast out of a conventional power plant coolant material (such as Li or Flibe) that can be used to absorb the heat from the fusion reaction, and also to breed tritium. Vacuum is nominally required only in the RTL, which can be pumped down before loading. The chamber itself would be filled with liquid or solid Li-bearing material with voids, chosen to mitigate the effects of the shock and neutron-induced damage to the first wall. The thickness of the liquid or solid fill would typically be greater than about 1 meter, which is sufficient to absorb the bulk of the neutron energy, provide a tritium breeding ratio above unity, and protect the first wall from neutron damage. The radius of the chamber would typically be in the range of 3 or more meters. Initial cost estimates for casting the RTL are 0.70 per shot, which is acceptable for a high-yield, low rep-rate IFE z-pinch power plant. Plans for initial feasibility tests of this RTL concept are presented.

*Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed-Martin Company, for the United States Department of Energy Under Contract DE-AC04-94AL85000.

Presenting Author: C.L. Olson

Plans for liquid lithium wall experiments in CDX-U*

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A flowing lithium first wall may lead to a significant advance in fusion reactor design, since it has the potential to eliminate the difficulties with power density and erosion, tritium retention, and cooling associated with solid walls. Lithium wall conditioning experiments have already been performed in tokamaks, and lithium sputtering and erosion tests have also been conducted in linear plasma devices. To address the interaction of an extensive liquid lithium plasma-facing surface with a toroidal plasma, a new set of experiments will begin this year in the Current Drive eXperiment – Upgrade (CDX-U) spherical torus (ST). The CDX-U plasma is intensely heated and well-diagnosed, and the planned studies will be the first of their kind in a spherical torus (ST). Since CDX-U is a modest ST, much smaller quantities of lithium are required to produce a toroidal liquid lithium plasma target than in large aspect ratio devices at comparable power densities. Lithium target designs and implementation plans for CDX-U will be presented.

*Work supported by US DOE contract #DE-AC02-76-CH03073.

Presenting Author: R. Majeski

TuC1

Calculations & Measurements of RMF Current Drive in FRCs

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Rotating Magnetic Fields (RMF) have been demonstrated to drive currents in many rotamak experiments, but use with an FRC confined in a flux conserver imposes special constraints. The strong current drive force results in a near zero density at the separatrix, and the high average beta condition requires the current to be carried in a relatively thin edge layer near the separatrix. The RMF can only penetrate into this layer by driving the azimuthal electron velocity synchronous with the RMF frequency. Build-up or maintenance of the flux throughout the FRC occurs due to an extreme flattening of the axial magnetic field profile near the field null, so that the small amount of RMF that penetrates that far can just overcome remaining resistive losses. The RMF cannot penetrate much beyond the field null, and current is maintained on the inner flux surfaces by an inward radial flow. Particle balance is maintained by a swirling axial flow and overall inward flow from inner to outer field lines. This process is seen using a new numerical code, and the resultant flux build-up and calculated profiles are demonstrated on the STX-RMF FRC formation experiment. code, and the resultant flux build-up and calculated profiles are demonstrated on the STX-RMF FRC formation experiment.

Presenting Author: K.E. Miller

TuC2

Numerical Study of Tilt Mode Stability in Field-Reversed Configurations

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The gross stability of the Field-Reversed Configurations (FRC) has been investigated using both MHD and hybrid (fluid electrons and kinetic ions) simulations. In contrast to previous work, the delta-f method is utilized to reduce numerical noise in the particle simulations. The stability properties of prolate (E>1) and oblate (E<1) configurations have been examined. For an oblate FRC, it has been shown that all n=1 modes (interchange, external tilt and radial shift modes) can be effectively stabilized in the MHD regime when the pressure profile is peaked enough and a conducting wall is used to stabilize the tilt mode.

The stabilizing effects of velocity shear and kinetic effects on the n=1 tilt mode in prolate FRCs have been studied. The linearized hybrid simulations show that there is a reduction in the tilt mode growth rate when s/E<1, but no absolute stabilization has been found for s/E values as small as 0.1, where s is the approximate number of ion gyroradii between the field null and the separatrix, and E is the separatrix elongation.

The preliminary results of nonlinear hybrid simulations at low values of s indicate that the tilt instability may saturate nonlinearly through the lengthening of the initial equilibrium and modification of the ion distribution function. These saturated states have been calculated to exist for many Alfven times (>10), maintaining field reversal.

Presenting Author: E. V. Belova

TuD1

Recent Liner Experiments Confirm Plans for Magnetized Target Fusion

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The premise of Magnetized Target Fusion is that fusion gain can be produced inexpensively using "liner" compression of a magnetized plasma. For adiabatic compression of a Field-Reversed Configuration (FRC), 10:1 radial compression gives impressive heating: 10 keV from an initial temperature of 250 eV. Recent experiments at the Air Force Research Laboratory will be described that have demonstrated how a 10-cm-diameter 1-mm-thick aluminum cylinder can be imploded using shock-free electromagnetic forces. The dimensions and energy density (liner velocity of 4 mm per microsecond) are the same as needed for interesting FRC compression. Although 10:1 radial compression implies considerable elastic-plastic deformation and thickening of the liner, experiments indicate highly symmetric deformation, and the inner surface of the metal remains a smooth cylinder according to an array of impact probes and side-on radiography. The time history of the radial implosion inferred by measuring the compression of a small magnetic field agrees with computer models of the implosion. Some further development of the liner implosion technique is necessary before liner-on-plasma experiments begin, but for the immediate future emphasis in MTF research has shifted to preparation of a suitable plasma target. Using models based on previous FRC experiments, LANL and AFRL are designing an experiment at Los Alamos that can generate a 5T pulse of magnetic field in 2.5 microseconds, giving FRC density ~ $10^{17}cm^{-3}$ and T ~ 250 eV. The FRC dimensions (separatrix radius ~ 2.5 cm and length ~ 30 cm) allow the FRC after formation to be translated and trapped in a liner close to the same size as that used for the liner implosion experiments. Diagnostics for FRC experiments will include a magnetic probe array to measure separatrix radius vs. axial position, side-on interferometry for density (and temperature by pressure balance), optical imaging for global stability, Thomson scattering for T_e (and density independently), and spectroscopy for impurity content.

Presenting Author: R. E. Siemon

Pulsed High Density Fusion

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Based on results from recent FRC acceleration experiments, together with the confinement scaling observed in past FRC experiments, a method has been determined by which an FRC can be compressed to high density and brought to ignition conditions in a rapid, repetitive manner. This regime is referred to as the Pulsed High Density (PHD) regime of MFE. Unlike MTF, the upper boundary of this regime remains below the density limit imposed by material strength limitations from the confining field.

Considerable data has been accumulated from various FRC experiments that span over two orders of magnitude in density and an order of magnitude in radius. Given the observed scaling with size and density, the required radius at a density of 10^{24} m^{-3} for a DT fusion burn with a gain > 1 is found to be ~ 1 cm.

The method by which the plasma density and temperature can be brought to fusion conditions is to start with a much simpler, low voltage FRC plasma source that can be repetitively pulsed. The energy necessary for burn is transferred to the FRC in the form of translational energy, which is produced by an inductive magnetized plasma accelerator (IMPAC) that is also capable of repetitive pulsing. The simplicity of this approach to fusion lies in the fact that the directed energy of the FRC mass, E_d (= $1/2Mv_{FRC}^2$) is much greater than the FRC internal energy (3/2NkT). Since E_d is in the form of a coherent translational motion, the confining magnetic fields, as well as accelerating fields, need to be no greater than required to contain the low-pressure FRC generated in the source coil (~ 0.4 T). This leads to a tremendous reduction in magnet mass as well as stored energy requirements for the accelerator. The conversion of the FRC directed energy into thermal energy occurs only after the FRC has reached the burn chamber where the FRC is slowed and compressed to fusion conditions. This chamber, which has a high magnetic field (~ 30 T), can be steady state and can thus generated by a superconducting magnet. The goal of concept exploration experiment would be the construction of an IMPAC device capable of producing a FRC plasma where all key parameters can be brought to within an order of magnitude of that required for a Q~1 fusion burn.

The PHD fusion envisioned also provides for a simple direct conversion of the plasma into directed thrust. Of all fusion reactor embodiments, only the magnetically confined plasma in the Field Reversed Configuration (FRC) has the linear geometry, low confining field, and high plasma pressure required for the direct conversion of fusion energy into high specific impulse and thrust, and would thus have direct applicability to deep space flight.

Presenting Author: J.T. Slough

TuD3

A concept of a local spherical blanket applied for MTF systems

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A concept of a local spherical blanket for pulsed fusion systems has been analyzed in great deal of detail by B.G. Logan ([1] and references therein) and, more recently, by M. Derzon [2], and D. Ryutov and A. Toor [3]. The blanket material (Li, LiPb, Flibe, or LiH) forms a sphere in the center of which a fusion energy release occurs. The fusion neutrons propagate through the blanket and breed tritium; they cause also the pulsed volumetric heating of the blanket, its evaporation and ionization. The ionized material serves then as a working fluid for the MHD generator.

We discuss a possible use of a spherical blanket for Magnetized Target Fusion (MTF) systems. These systems can be driven by the current pulse with the pulse duration in the range of a few microseconds and a maximum current ~ 10 MA. The current pulse with such parameters can be generated by crushing magnetic flux conserver (integrated with the spherical blanket) by a fast projectile [3, 4]. A power transmission line will be integrated with the spherical blanket.

In each shot, the ball made of one of the aforementioned materials, together with the enclosed fusion capsule, will be dropped into the reactor chamber. We consider design options oriented both at a significant ionization of the ball, and at its mere evaporation. In the latter case the vapor will be used to drive a turbo-generator. We discuss electrical and mechanical issues of the spherical blankets and conclude that, potentially, they can lead to an attractive MTF power reactor.

Work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract W-7405-ENG-48.

1. B.G. Logan. "Inertial Fusion Reactors Using Compact Fusion Advanced Rankine (CFARII) MHD Conversion," Fus. Eng. And Design, v. 22, p. 1953 (1993).
2. M.S. Derzon. "Development of a Low-Technology Fusion reactor," Paper presented at the Z-Pinch IFE Workshop, Albuquerque, NM, April 27-28, 1999.
3. D. Ryutov, A. Toor. "Stand-off Energy Sources for Z-pinch Implosions," UCRL-ID-135082, July 1999.
4. R.P. Drake, J.H. Hammer, C.W. Hartman, L.J. Perkins, D.D. Ryutov. "Submegajoule liner implosion of a closed field line configuration." Fusion Technology, v.30, p.310 (1996).

Presenting Author: D.D. Ryutov

TuE1

Penning Fusion eXperiment – Ions (PFX-I)

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The Penning Fusion eXperiment – Ions (PFX-I) is a unique magnetic confinement concept based upon the traditional Penning trap. Present work is concentrated on producing a uniform, spherical, electron plasma, which will then provide confinement and spatial focussing for positive ions. We are developing an electron diagnostic based upon the Stark splitting of Hydrogen emission lines when neutral H_2 gas is added to the electron cloud confined in the trap. For our experimental conditions ($n_e > 10^{10}$ cm^{-3}), the H_α π-lines should be separated by more than an angstrom. A second diagnostic, which involves the destructive dumping of the trapped cloud, has also been developed to study the population and energy distribution of the electrons. Initial results of PFXI operation at applied voltages up to 2kV and magnetic fields to 2T will be presented. Electron inventory, particle and energy confinement times, and energy distributions are reported. Initial results of gas injection experiments will also be presented.

Presenting Author: K.R. Umstadter

Confinement Of High Density Pure Ion Plasma In A Cylindrical Trap

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A novel method for containing a pure ion plasma at densities that are relevant to thermonuclear fusion has been modeled. The method combines the confinement principles of a Penning-Malmberg trap and a pulsed theta-pinch.

Starting with a low-density trapped deuterium plasma at the Brillouin limit, the field is ramped to as high an intensity as practicable. With only one-component in the plasma, transport losses are very low, i.e., the conductivity is high, the magetic diffusion time is long, and the ramped field does not penetrate the plasma. Simultaneously, ions are injected to increase the density of the target plasma, and the mutual electrostatic repulsion opposes the inward pinch. However, without a magnetic field inside the surface of the plasma, the ions drift outward until a balance is established between the outward driving forces (centrifugal, electrostatic, pressure gradient) and the inward J x B force. The result is trapping of the ions in a current sheet with local densities much higher than are typically associated with the Brillouin limit. An equilibrium calculation has been made to illustrate the configuration. Using a relativistic, 1-D, isothermal fluid model, the plasma can shown to maintain force balance in a hollow, 49-cm diameter, 0.2-cm thick cylinder whose density exceeds 4×10^{14} cm^{-3}. A 2d-3v, relativistic, electromagnetic, particle-in-cell numerical simulation is being adapted to investigate the velocity equilibration, particle injection efficiency, and maximum energy yield.

Presenting Author: S.F. Paul

TuE3

Stability of Virtual Cathodes for the Periodically Oscillating Plasma Sphere (POPS)

R. A. Nebel, J. M. Finn

Recent theoretical work[1,2] has suggested that a tiny oscillating ion cloud may undergo a self-similar collapse that can result in the periodic and simultaneous attainment of ultra-high densities and temperatures. Theoretical projections[1] indicate that such a system may have net fusion gain even for an advanced fuel such as D-D. Schemes have also been suggested where a massively modular system consisting of tens of thousands of these spheres can lead to a very high mass power density device (comparable to a LWR).[1] Such systems should be very economically competitive.

However, a major uncertainty in this plasma system is the behavior of the electrons. In this paper we will show that the required electron cloud is susceptible to an instability which is analagous to the Rayleigh-Taylor mode present in fluid mechanics.[3] Linear stability analysis indicates that the mode has marginal stability limits at the Brillouin density limit and in the limit as the ratio of the effective Debye length to the plasma radius goes to inifinity. Fortunately, solutions of the marginal stability profiles indicate that stable profiles which are sufficiently close to those required for POPS do exist as one approaches the Brillouin limit as well as in the large Debye length to radius limit.

In the kinetic limit (counter-streaming electrons) one expects this instability to go over to a type of two-stream instability. Simple two-stream analysis suggests that the mode may be absolutely stable at a finite value of ratio of the Debye length to plasma radius. This conclusion is consistent with previous experimental results[4] which showed that large scale oscillations occured in gridded electron IECs when the current exceeded a critical value. Our theory correctly predicts the observed scaling of the critical current with the voltage. It is also consistent with the observed scaling of the oscillation frequency with voltage.

Unfortunately, these previous experiments did not include profile measurements. Consequently, it isn't clear exactly where these marginal points occurred with respect to the important parameters of the Debye length and the electron distribution function. Determining these values is critical since the required applied voltage for a POPS system scales like the Debye length squared. We are proposing to repeat these experiments (as well as a general study of equilibrium and stability of virtual cathodes) with detailed profile measurements. We plan to carry out these measurements on an existing gridded IEC device, the Intense Neutron Source.[5] The machine will be described and these proposed experiments will be discussed.

1. R. A. Nebel, D. C. Barnes, Fusion Technology 38, 28 (1998).
2. D. C. Barnes, R. A. Nebel, Physics of Plasmas 5, 2498 (1998).
3. J. M. Finn, R. A. Nebel, to be submitted to Physics of Plasmas.
4. R. L. Hirsch, Physics of Fluids 11, 2486 (2000).
5 R. A. Nebel, A. J. Cole, K. R. Umstadter, proceedings of the 18th IEEE Symposium on Fusion Engineering, Albuquerque, October (1999).

Presenting Author: R. Nebel

WeA1

Fast-Ignition Inertial Confinement Fusion Research*

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The fast-ignitor concept is an innovative approach to achieving break-even and gain in inertial confinement fusion experiments. In this scheme, an ultra-intense short-pulse laser beam is used to ignite a pre-compressed mass of dense thermonuclear fuel. Significant research effort will be required to demonstrate sufficient control of the ignition process to make this scheme feasible, and difficult issues include laser channel formation in the low-density plasma atmosphere surrounding the fuel, efficient energy conversion into relativistic electrons with the appropriate spectrum, and collimated transport of the electron beam into the main fuel. The substantial potential benefits of success include higher target gains, reduced driver energy requirements, and relaxed target fabrication requirements. We describe recent progress in fast-ignitor research at LLNL using the Petawatt Laser Facility, and we discuss plans for future collaborative research between LLNL, General Atomics, University of California/Davis, Osaka University, and Gesellschaft für Schwerionenforschung/Darmstadt.

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Presenting Author: J. A. Koch

Compression of High Power Laser Pulses in Plasma

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While achievable laser intensities have grown remarkably during recent years mainly due to the method of chirped pulse amplification (CPA), using CPA to attain even higher powers would demand suitable gratings for handling very high power and very high total energy. For further progress yet, it would be necessary to develop as well lasers, amplifiers, and gratings operating at wavelengths shorter than a micron. Yet plasma is an ideal medium to form a grating capable of processing very high power and very high total energy. We have in mind compression of powers almost to exawatts per square cm or fluences to kilojoules per square cm, prior to the vacuum focus. For energy applications, pulse compression does not really need high fidelity within each frequency range. Thus, plasma is ideal for applications for delivering high power, which include the fast igniter, but really cover a much broader range of possibilities.

One plasma based compression mechanism involves a so-called "superradiant" effect, where a long pump pulse is depleted by a short counterpropagating pulse; here the nonlinear interaction of the plasma electrons with the lasers dominates the plasma restoring motion due to charge imbalances [1]. Amplification in this regime is not very sensitive to the frequency detuning between the two lasers. A more straightforward method of amplification and compression is simply by fast backward Raman scattering, where the amplification process outruns deleterious processes associated with the ultraintense pulse [2]. Here the frequency detuning between the two lasers can be chosen precisely equal to the plasma frequency. In this regime, it is possible to employ a very interesting nonlinear filtering effect, which makes possible highly efficient amplification of a moderate seed-pulse, while suppressing small noise [3].

[1] G. Shvets, N. J. Fisch, A. Pukhov, and J. Meyer-ter-Vehn,
"Superradiant Amplification of an Ultra-Short Laser Pulse in a Plasma by a Counter-Propagating Pump," Phys. Rev. Lett. 22, 4879 (November, 1998).
[2] V. M. Malkin, G. Shvets and N. J. Fisch,
"Fast compression of laser beams to highly overcritical powers,"
Physical Review Letters 82, 4448 (1999).
[3] V. M. Malkin, G. Shvets and N. J. Fisch,
"Detuned Raman Amplification of Short Laser Pulses in Plasma",
to appear in Physical Review Letters (February 2, 2000).

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Advances in IFE Targets for Heavy Ion Fusion*

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Two-dimensional, integrated, Lasnex [1] calculations of a close-coupled version of the distributed radiator, heavy ion target predict gain of 130 from 3.3 MJ of beam energy. To achieve these results, the case-to-capsule ratio was decreased by about 25% from our previous targets [2]. Reducing the driver energy required should reduce the cost of the driver and, ultimately, the cost of electricity. The smaller hohlraum results in smaller beam spots than had been previously assumed; this puts renewed emphasis on controlling emittance growth in the accelerator and on space-charge neutralization in the reactor chamber. The close-coupled heavy ion target also opens up the possibility of a high gain Engineering Test Facility from a 1.5-2 MJ driver; calculations predict that gain of 90 is achievable from 1.75 MJ of beam energy. This could have a significant impact on the development cost of heavy ion fusion since the same accelerator could be used to study low gain targets and high gain targets. Given adequate repetition rate, the same accelerator could even be used to drive multiple chambers for a demonstration power plant. In addition to describing our close-coupled targets, we will present work-in-progress on target concepts that relax the requirements on beam spot size and/or beam pulse duration.

[1] G. B. Zimmerman, W. L. Kruer, Comments on Plasma Physics and Controlled Fusion, 2, 51 (1975).
[2] M. Tabak, D. Callahan-Miller, Phys. Plasmas, 5, 1895 (1998).

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Presenting Author: D.A. Callahan-Miller

WeB1

The Flow-Stabilized Z-Pinch Experiment: ZaP

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The stabilizing effect of an axial flow on the $m=1$ kink instability in z-pinches has been studied numerically by reducing the linearized ideal MHD equations to a one-dimensional eigenvalue equation for the radial displacement. The principal result reveals that a sheared axial flow stabilizes the kink mode when the shear exceeds a threshold value which is inversely proportional to the wavelength of the mode. Nonlinear simulations support the stabilizing effect.

The implications of this stabilizing effect are investigated with a flow-through Z-pinch experiment, ZaP. The experiment produces a Z-pinch plasma which is 50 cm in length by initiating the plasma with a one meter coaxial gun. The evolution of the plasma's magnetic structure is measured with surface mounted magnetic probes which form an axial array and two azimuthal arrays. The azimuthal arrays allow the measurement of the fluctuation levels of the azimuthal modes $m=0,1,2,3$. Velocity is measured by calculating the Doppler shift of a C-III line using a spectrometer. Time-dependent, single-point density measurements are made using a HeNe laser interferometer. Optical images are obtained using a fast framing camera. An overview of the experimental program and results will be presented

This work is supported by the Department of Energy.

Presenting Author: U. Shumlak

Staged Z-Pinch for Controlled Fusion

P. Ney, H.U.Rahman
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and
F. J. Wessel, and N. Rostoker
University of California, Irvine

A staged Z-pinch is considered in which an annular plasma shell made of a high Z material, like Kr, implodes onto a coaxial plasma target made of a low Z material, like deuterium or a deuterium-tritium mixture. The target plasma could be initiated either by exploding a cryogenically extruded fiber or by filling the annular shell with a gas puff or plasma puff. Modeling is performed with a 2D radiation-MHD code. A parameter study is made to determine the sensitivity of this configuration to initial conditions of the shell and the target plasmas. An axial magnetic field is essential for a stable implosion and efficient energy coupling to the final load. Using a 50-50 mixture of deuterium-tritium as a target, the fusion energy gain is optimized by adjusting the initial parameters. Breakeven conditions are computed for the nominal parameters characteristic of the 50 kJ UCI Z-pinch facility.

Presenting Author: H.U. Rahman

Practical Applications of Helicity to ICCs.

S. Woodruff, D.N.Hill, E.B.Hooper, B.W.Stallard

The concept of helicity (flux-linkage) has a wide variety of applications – from laboratory- to astrophysical-scale plasmas. This talk is intended to survey the subject of helicity from an experimentalist's perspective and show how helicity may be employed as a useful tool for many different confinement concepts. Particular emphasis will be given to the practical application of this theory with examples from SSPX, SPHEX, HIST, HIT-II, and NSTX. Comparison with traditional energy balances will be given, and it will be shown that helicity considerations are complimentary, and in many instances entail a simpler approach to predicting plasma evolution and equilibrium.

The original application of helicity in the RFP and spheromak [1] has been broadened to entail partially relaxed systems such as low-aspect ratio helicity injected tokamaks and FRC formation. Devices such as HIT-II and NSTX successfully employ DC helicity injection schemes to augment conventional ohmic current drive. FRC's operate in the limit of zero helicity content and may be formed by the merging of counter-helicity spheromaks. There is evidence that this annihilation of helicity may provide free energy for ion heating.

Future devices are designed to fully exploit the concept of helicity: SIHI [2] - a recently proposed device will inductively sustain a CT by constant supply of helicity, omitting electrodes; and SPHERA [3] will form tokamaks by helicity injection with toroidal field supplied by a plasma column.

Despite decades of work, it is still not clear how helicity injection serves to sustain the configuration, although many theories exist for the experimentalist to test. These include (amongst others) the MHD dynamo, the 'tangled discharge model' [4], multiple plasmoid coalescence [5], and a proposal related to rotating magnetic fields [6]. An overview of these will be given.

[1] J.B.Taylor, Rev. Mod Phys. 58, 741 (1986)
[2] T.R.Jarboe, Fus. Tech., v36 (1999)
[3] F.Alladio et al, Bull. Am. Phys. Soc. 41, 1405 (1996)
[4] M.G.Rusbridge, Plasma Phys. 19, 499 (1977)
[5] M.Nagata, Proc. Eur. Phys. Soc. V22C, 807 (1998)
[6] T.R.Jarboe Phys. Plasmas v5, no5, 1807 (1998)

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Presenting Author: S. Woodruff

WeC2

Overview of Spheromak Formation Experiments on SSPX

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We present results from spheromak formation experiments carried out during the first nine months of operation of the Sustained Spheromak Physics Experiment (SSPX) at Livermore. The SSPX experiment was built to study energy confinement and transport in spheromak plasmas sustained by DC coaxial helicity injection. The SSPX plasma is confined within a 1m dia. copper flux conserver which is designed to handle maximum edge poloidal magnetic fields of 1.5T produced by I_{tor}=1.2MA discharges. Unlike previous spheromak experiments, SSPX does not have a transition region isolating the injector from the spheromak, and it has a large-radius injector ($r_{inj}/r_{fc} = 0.7$) with radial magnetic fields to increase the drive efficiency and minimize field errors. Peak edge fields of 0.3T and toroidal currents up to I_{tor}=500kA have been obtained.

Initial spheromak formation experiments have examined ejection thresholds, buildup of stored energy, and particle control. Our results show that the unique radial field geometry in the coaxial region produces a lower threshold current for ejecting spheromaks into the flux conserver than initially expected. Furthermore, the normalized threshold current density, $\lambda=j/B=\pi/\Delta$, depends on the field strength B rather than just the gap width Δ, which means that we capture less flux in the spheromak than simple estimates yielded.

The initial buildup of spheromak currents and fields in SSPX continues until the formation bank runs out of energy. We have modeled this buildup by considering both global energy and helicity balance. So far, the coupling efficiency from the coaxial source to spheromak plasma $\varepsilon=(dW_{sph}/dt)/P_{bank}$, has reached 28%, compared to the 50% efficiency expected for the SSPX configuration. We expect this to increase as the plasma gets hotter. We have used these initial data to predict SSPX performance under a variety of operating conditions (varying Te and magnetic field decay time) when we start to use the 1.5MJ sustainment bank to extend the pulse length.

In parallel with carrying out the formation experiments, we have been working to improve the capability of the SSPX device. We now have density measurements from a multichord CO2 interferometer from LANL, and we are testing an ultra-short-pulse microwave reflectometer for measuring the density profile. Later this year we plan to have Thomson scattering operating routinely. We will also finish installing a set of bias field coils to significantly change the magnetic geometry to allow flux-core spheromak formation, which should have a substantially reduced threshold current.

*The SSPX Team includes members from LLNL, LANL, SNL, UC Berkeley, UC Davis, Univ. of Wash., Univ. of Wisconsin.

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Presenting Author: D.N. Hill

WeC3

Formation and Self-Similar Expansion of a Spheromak Configuration in the Absence of a Flux-Conserving Boundary

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We have investigated spheromak formation for the situation where the plasma is not bounded by conducting walls. Isolation from walls is achieved by using a vacuum chamber that is an order of magnitude larger than the gun. This arrangement not only eliminates plasma-wall interactions but also provides excellent viewing of plasma dynamics. The evolving configuration is photographed using a pair of high-speed, intensified, gated CCD cameras and is also diagnosed with a 3D magnetic probe array. Four distinct operational regimes are identified for various combinations of gun current and bias. The ratio of gun current to bias flux is called the gun lambda, while in the plasma the ratio of force-free current to magnetic field is called the plasma lambda. For gun lambda slightly below the threshold for spheromak formation, photos show visibly twisted structures and magnetic measurements indicate that no detachment takes place. In contrast, for gun lambda above the threshold for spheromak formation, (i) photos indicate the plasma has an approximately axisymmetric toroidal shape, (ii) magnetic probe data clearly demonstrate a force-free helical magnetic field structure propagating away from the gun, (iii) toroidal and poloidal fluxes calculated from the measured fields indicate substantial poloidal flux amplification and associated reduction in toroidal flux, (iv) the spatial dependence of the plasma lambda is consistent with the profile expected for a decaying isolated spheromak, and (v) the time evolution of plasma lambda indicates that the unbounded spheromak undergoes a self-similar, helicity conserving expansion.

Presenting Author: P.M. Bellan

WeD1

Active Feedback Control of the Resistive Wall Mode in HBT-EP*

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We report the first observation in a tokamak of the use of active feedback control to suppress the onset of the wall stabilized n=1 external kink mode destabilized by finite conductivity when the stabilizing wall is non-ideal, *i.e.* the resistive wall mode (RWM). In toroidal devices such as the reversed field pinch (RFP) or the advanced tokamak (AT) which rely on a nearby conducting wall to stabilize the current or pressure driven external low-n kink mode, the lifetime and/or beta limit of these devices is set by the onset of the RWM [1,2] which grows on the much slower time scale of the flux penetration through the conducting wall rather than the very rapid MHD Alfven time scale. These RWMs have been identified as limiting phenomena in the RFP [3,4] and in the AT [5,6] and similar phenomena are expected to be important in a wide range of toroidally confined plasmas including the Spherical Torus, Spheromak, and Field Reversed Configuration. One approach to the stabilization of these RWM instabilities is to use a network of active feedback coils configured so that the electrical response of the resistive wall simulates that of a perfect conductor. This so-called 'intelligent shell' or 'smart shell' was proposed by Bishop [7] and has been implemented in the HBT-EP tokamak with 30 independent sensor/driver feedback loops mounted behind a 2 mm stainless-steel resistive wall located near the plasma boundary. The time constant for flux soak through of this stainless-steel wall is about 300 microseconds which is consistent with the observed growth time of the RWM as expected from the theory. The performance of the HBT-EP smart shell feedback stabilization system has been modeled by a 3D finite element electromagnetic code, VALEN, and is in agreement with the observed stabilization of the RWM. The VALEN code has also been quantitatively benchmarked against the predictions of a large aspect ratio analytic MHD model.

* Work supported by US DOE Grant DE-FG02-86ER53222.

[1] D. Pfirsch and H. Tasso, Nuclear Fusion 11, 259 (1971).
[2] C. G. Gimblett, Nuclear Fusion 26, 617 (1986).
[3] B. Alper, M. K. Bevir, H. A. B. Bodin, *et al.*, Plasma Physics Controlled Fusion 31, 205 (1989).
[4] P. Greene and S. Robertson, Physics of Fluids B 5, 556 (1993).
[5] E. J. Strait, T. S. Taylor, A. D. Turnbull, *et al.*, Physical Review Letters 74, 2483 (1995).
[6] A. M. Garofalo, A. D. Turnbull, M. E. Austin, *et al.*, Physical Review Letters 82, 3811 (1999).
[7] C. M. Bishop, Plasma Physics Controlled Fusion 31, 1179 (1989).

Presenting Author: G. A. Navratil

Nonlinear Dynamics of Feedback Modulated Magnetic Islands

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IFS

Resistive instabilities (e.g., neoclassical tearing modes, error-field driven locked modes, resistive shell modes) play a limiting role in virtually all toroidal magnetic confinement devices (e.g., tokamaks, reversed field pinches, and stellerators). Direct feedback stabilization via externally imposed rotating magnetic perturbations is currently under serious consideration as a possible alleviation mechanism for troublesome resistive modes. Indeed, feedback systems of varying capabilities are currently installed on the HBTX tokamak, the DIII-D tokamak, and the MST reversed field pinch. Unfortunately, the present state of understanding regarding the dynamics of non-linear tearing modes under the influence of feedback perturbations leaves something to be desired. Previous treatments are marred by an excessively naive treatment of plasma viscosity: either this crucial effect is entirely neglected, or it is modeled in a crude and unsatisfactory manner. We present, for the first time, a comprehensive treatment of feedback modulated tearing mode dynamics with a realistic treatment of viscosity. We find three regimes of operation, depending on the modulation frequency. For slowly modulated islands, the perturbed velocity profile extends across the whole plasma. For strongly modulated islands, the perturbed velocity profile is localized around the magnetic island, but remains much wider than the island. Finally, for very strongly modulated islands, the perturbed velocity profile essentially collapses to a boundary layer on the island separatrix. We solve for the velocity profiles, the island equation of motion, and the island width modulation equation in each of these three regimes. A relatively simple set of equations which interpolate between all three regimes is obtained. We find that the ion polarization correction to the island width modulation equation, which has previously been reported to be stabilizing, is, in fact, destabilizing in all three regimes. We demonstrate that a previously reported experimental verification of the alleged stabilizing effect of the ion polarization term, from HBTX feedback experiments, can be accounted for by a different mechanism.

Presenting Author: R. Fitzpatrick

WeP1

'Stellerator-Pinch': A Composite Fusion Concept

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The concept of a new fusion device which is a composite of a stellarator and linear theta pinches is proposed. The device in a race track configuration consists of two linear segments which are theta pinches. The latter are joined via two halves of a stellarator which from the curved segments. The primary purpose of theta pinch sectors of modest lengths will be to provide a technologically convenient and efficient method of producing a high β fusion grade plasma in a stellarator, which remains a difficult issue. In view of the very high densities and moderately high temperatures available from theta pinches, one can envision a modest size closed device with $\eta\tau_{conf}$~$5x10^{14}$ cm^{-3} sec, which indicates fusion feasibility.

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NCSX Facility Design Progress

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In 1999, a point design for the National Compact Stellarator Experiment device was developed. Though not a complete conceptual design for the NCSX, the 1999 point design showed that a high-beta CS plasma configuration that satisfies physics requirements for plasma stability can be realized in a practical device implementation. A key feature of the 1999 design was its use of the existing toroidal and poloidal field coils from the former PBX-M tokamak device to provide a "coarse" background field. Alternative background coil options, including those with all-new coils optimized for NCSX instead of existing ones, are now being explored. Saddle coils and deformed TF and PF coils are options being studied for providing the "fine" corrections needed to produce the required fields with high accuracy. Since the NCSX device must provide access for required heating, diagnostic, and pumping systems in order to fulfill its physics objectives, access is an important criterion for evaluating options. Recent progress in the exploration of optimized coil systems and in the definition and evaluation of access requirements for physics tools will be reported.

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WeP3

Reduced edge instability and improved confinement in the MST RFP

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Reduction of core-resonant magnetic fluctuations and improved confinement in the MST is reliably achieved through control of the poloidal electric field. However, the achieved confinement has been limited by a burst-like instability originating in the plasma edge. Now, improved control of the poloidal and toroidal electric fields allows multi-ms suppression of this instability, along with core fluctuation reduction, leading to (1) a central electron temperature of 840 eV at 470 kA, (2) a total beta > 12 percent at 200 kA, (3) a steepened electron temperature profile, (4) a reduction of the edge temperature and density, and (5) a reduced electron thermal diffusivity. Equally important is that the estimated energy confinement time at 200 kA exceeds the "constant beta" scaling that has characterized the world RFP confinement database

Presenting Author: B.E. Chapman

WeP4

Pinch discharges in the Electric Tokamak

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Initial experiments in the Electric Tokamak (ET) indicate that low q (<0.5) discharges can be produced which are quite stable when the neutral density is high. Good confinement and burn out of the neutrals require a few hundred kilo amperes plasmas whereas the burn out in a tokamak plasma has been accomplished at 40 KA plasma current or even lower. High current pinch discharges are possible in ET using its full ohmic system. Results of the initial attempts in achieving burn out in a pinch discharge will be presented.

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Physics Issues for a Plasma Confined in a Dipole Field

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The levitated dipole concept, as embodied in LDX, was inspired by increased understanding of magnetospheric physics and it offers a path towards a novel confinement geometry for magnetic fusion. In the dipole approach the plasma is confined within the dipole field of a "floating ring". To avoid plasma loss to the ring supports the ring will be levitated within a large vacuum chamber.

In the levitated dipole concept the magnetic field is poloidal and the field lines are closed. Thus there is no rotational transform or shear and similarly to magnetospheric plasmas stability (in the region between the pressure peak and the vacuum vessel) derives from plasma compressibility. This imposes a constraint on the pressure gradient which results in the requirement of a small ring in a large vacuum chamber. It has been shown that, when the pressure gradient constraint is satisfied, there is no limit on equilibrium beta. Furthermore MHD ballooning modes have also been shown to remain stable at large beta (since at high beta the magnetic flux spreads out at the midplane there is a geometric constraint on peak pressure due to having a finite size vacuum chamber).

The closed field line geometry introduces the possibility of convective cells and convective flows. Such flow patterns can result from non-axisymmetric heating or fueling. Importantly, the convective flows which move particles do not necessarily transport energy. In fact for a marginal stable pressure gradient such flows are adiabatic and do not transport energy and for a sub critical gradient the flows would transport energy inwards. Therefore convective flow patterns could provide an unorthodox method for the heating and fueling of a dipole confined plasma.

Since MHD stability limits the pressure gradient it also sets a limit on the diamagnetic drift frequency ($\omega_* \propto dp/dR$), i.e. $\omega_* \leq \omega_c$ with ω_c curvature drift frequency ($\omega_c \propto T/Rc$). It has been shown that with this ordering drift waves tend to be stable. Therefore a dipole confined plasma can theoretically be stable to the drift waves that are a source of micro-turbulence in many confinement devices.

Presenting Author: J. Kesner

WeP6

Superconducting Morozov's Ring for refuelling and controlling plasma profiles

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The Morozov transporter ring represents a current carrying metal ring moving across the high temperature plasma volume while spending only a finite time (1-100 ms) inside the plasma. Inside the plasma the ring is magnetically insulated from the plasma electrons and is electro-statically insulated from the ions (by the electric charge delivered to the surface of the ring by the most energetic ions or alpha-particles). Depending on the magnetic flux, the local magnetic configuration near the ring represents either FRC or levitron with an inverse (and, thus, MHD stable) plasma pressure gradient localized at the separatrix. Inside the separatrix, the confinement of positive energetic) particles represents a new, interesting mixture of magnetic confinement with the strong electrostatic effects. Both warm (high speed) Morozov's rings and Super-conducting (low speed) rings can be possibly developed for many applications.

The Super-conducting Morozov's Rings (SMR) is the most clean implementation of the Morozov idea of an autonomous object inside the high temperature plasma. Being non-perturbative for the plasma, SMRs can significantly enhance the diagnostics capabilities of the present day mid-size or large tokamaks or stelalrators (NSTX, JET, JT-60, Tore-Supra, LHD, etc) which have a significant portion of hot ions or alpha-particles. In addition, SMRs can be used as a transporter of the plasma fuel, which can be released locally at any desirable point inside the plasma. This opens unique opportunities for controlling the plasma profiles and for the transport studies.

SMRs could be especially important for the JET D-T experiments where the fusion alpha-particles can provide an extreme in the electro-static screening of ions. Also, D-T refuelling with SMRs might free the neutral beam lines from contamination by the tritium.

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Warm Morozov's Rings for the helium exhaust from the fusion reactors

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While the super-conducting Morozov's Rings (SMR, see a separate Abstract for ICC2000) is the most straightforward implementation of the Morozov idea of an autonomous object inside the high temperature plasma, their application for fusion reactors is unlikely because of the volumetric heating from the neutrons and gamma radiation. On the other hand, the development of warm Morozov's rings (WMR) with the normal conductivity may open the opportunities for resolving such fundamental issues as refueling and controlling plasma profiles even in the fusion reactor environment. WMR typically requires a high speed in the range of 1 km/sec. For tokamaks and stellarators it can be reduced, if the trajectory of the ring directs into higher magnetic field component, perpendicular to the plane of the ring.

Here, I suggest an additional possibility of using WMRs to solve the helium exhaust problem in the fusion reactors.

The standard Morozov's rings are magnetically insulated from the plasma electrons while being quickly positively charged by unmagnetized alpha-particles and then electro-statically insulated from the alpha-particles and ions. The electric field of the ring is localized inside the separatrix of the magnetic configuration which holds an excess of electrons. It is possible to electrically connect (e.g., with a thin resistive metal wire) this electron layer with the surface of the ring and, thus, to allow a continuous current of electrons, and as a result, the alpha-particles to the ring. Then, if the ring can retain, at least temporarily (for a transit time of the ring), neutral He gas, then alpha particles with the energy above a certain value will be partially removed from the magnetic surfaces within the trajectory of the ring.

The magnetic flux and the geometry of the ring can be adjusted to differentiate between magnetized and un-magnetized alpha-particles at different energies. In combination with some other (plasma physics) means of transporting cooled, but not thermalized, alpha-particles from the central core to the periphery of the reactor, multi-WMRs can clean up all the alpha particles on the outer surfaces before the alpha-particles are getting mixed up with the thermal population of the plasma ions.

From the physics point of view, this WRM scheme differs significantly from the standard approach, relying on a divertor for the helium exhaust, when the alpha-particles are allowed to be mixed with the bulk of the plasma. This makes them practically indistinguishable from the T or D ions. Instead, the WRM scheme uses the big initial separation in energy between the fusion alphas and the plasma particles.

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WeP8

Single-Particle Heating by Rotating Magnetic Fields in FRCs Having Closed Flux Surfaces

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Rotating magnetic fields (RMFs) have been successfully used, particularly in rotamak devices,[1,2] to make plasmas, drive toroidal currents, and obtain field reversal. The standard (i.e., even-parity) RMF configuration, however, is predicted to open the FRC's field-line structure.[3,4] Larger, higher power RMF experiments are in progress.[5] These aim to produce higher temperature plasmas, more susceptible to open-field-line particle and energy losses. This possibility strongly motivates the studies of field-closure-conserving RMF configurations and their effects on particle confinement and heating. Recently, static odd-parity transverse fields have been shown to preserve field closure in FRC's.[6] Odd-parity RMFs are also predicted to heat ions[7] and electrons[8]. In this paper, we shall describe an experiment to study parity and particle heating in a small FRC. The experiment would compare operation of the same plasma device with both even-parity and odd-parity RMFs. Due to its small volume, the experiment would operate at high power/unit volume at low power levels. Combined with a low neutral pressure, detrimental plasma-wall interactions and atomic physics effects would be minimized. Numerical modeling predicts that 10 kW of odd-parity RMF power coupled into a 1-cm radius, 10-cm long plasma column, will produce an FRC with $T_e \sim 100$ eV, $n_e \sim 10^{13}$ cm^{-3} at a field of 200 G and $S \approx 1$.

Presenting Author: S.A. Cohen

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[1] H.A. Blevin and P.C. Thonemann, Nucl. Fusion: Supplement, Part 1, 55 (1962).
[2] For a recent review, see I.R. Jones, Phys. Plasmas 6, 1950 (1999).
[3] M.S. Chance, J.M. Greene, and T.H. Jensen, Geophys. Astrophys. Fluid Dynamics 65, 203 (1992).
[4] P. Watterson, J. Plasma Physics 46, 271 (1991).
[5] Slough and Miller, to appear in Phys. Plasmas
[6] Cohen and Milroy, submitted to Phys. Plasmas
[7] Cohen and Glasser, in preparation
[8] Glasser and Cohen, in preparation

WeP9

Translation, Confinement and Current Drive of Field Reversed Configuration in TCS

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The ultimate goal for the TCS experiments is to demonstrate current drive and flux sustainment of a hot FRC in a steady state by applying externally a rotating magnetic field. FRCs have been successfully produced, translated and captured in a separate, low field confinement chamber. The density is reduced from about 10^{21} m^{-3} to 10^{20} m^{-3}, which is more suitable for the RMF current drive.

The FRC is produced by the standard field-reversed theta pinch formation method in the formation chamber, then accelerated (at $\sim 3\times10^5$ m/s) into the confinement chamber. After reflection by a 10:1 magnetic mirror at the end of the confinement chamber, the directed energy is thermalized and the total temperature (T_i+T_e) increases by an order of magnitude, up to ~500 eV. The lifetime of the translated FRC is better than predictions based on high density empirical scaling, but is still severely limited by impurities picked up after reflection from the rear mirror. Measurements from both the bolometer and spectroscopy show a strong correlation between the flux decay time and impurity radiation. Octopole field coils are now being installed on the rear cone to minimize wall contact.

The RMF current drive system has been installed on TCS and is presently being tested. The RMF power supply (constructed at LANL) consists of two, timed tank circuits (40 kV, 6kA in each circuit), which produces a 50 G rotating magnetic field with a frequency of 159 kHz. Initial results from the RMF experiments will be reported.

Presenting Author: H.Y. Guo

WeP10

An MHD Model of Rotating Magnetic Field Current Drive in an FRC

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The use of a Rotating Magnetic Field (RMF) as an electron current drive mechanism, to form and sustain a field-reversed configuration (FRC), has been studied numerically. Previous models that were used to study the penetration of an RMF have not included fluid flow or pressure equilibrium. Here these results have been extended through the development of a fully two-dimensional (r-theta) MHD model. The model has been applied to two classes of problems: 1) For the formation problem, an RMF is applied to a plasma column with an initially uniform background axial magnetic field and density. The RMF induced current reverses this bias field, forming an FRC. 2) For the sustainment problem, an RMF is applied to a pre-existing FRC. Here the code employs an option to include some three-dimensional effects to satisfy the average beta condition and equalize pressure and density between inner and outer field lines.

Preliminary results are encouraging. The model predicts the formation of an FRC from a uniform background density on a timescale comparable to experimental observations in the STX experiment. In addition, application of the model to a pre-existing FRC shows that an equilibrium can be found with only partial penetration of the RMF. This produces a current profile that peaks near the FRC edge and a magnetic field profile in good qualitative agreement with experimental observations.

Presenting Author: R. D. Milroy

WeP11

Modeling High Density Field Reversed Configurations

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Magnetized Target Fusion (MTF) could achieve fusion conditions by compressional heating of a magnetized target plasma, such as imploding a Field Reversed Configuration (FRC). We are modeling these large density and temperature FRC's similar to those of the early theta pinch days using a variety of approaches. A physics based semi empirical scaling law[9] model is used to benchmark[10] our designs against the existing (but limited) world database. The two dimensional MagnetoHydroDynamic (MHD) fluid code MOQUI[11] is also being used to validate the time evolution of non ideal MHD equilibria during formation, compression, heating and translation. Other particle fluid hybrid models are being built to investigate the non MHD electron response to toroidal precession by eliminating the usual assumptions that tie electrons to magnetic field lines. The goal of the modeling effort is to extend our predictions to the MTF regime of interest ($n > 5\text{x}10^{16}\text{cm}^{-3}$, $T_e+T_i > 500\text{eV}$. We have identified several key FRC physics and MHD issues, including stability, flux retention, scaling into the collisional regime, and scaling to large density. The results are guiding us in this year's design and fabrication of the FRC physics experiment at Los Alamos National Laboratory with AFRL participation.

1. R Siemon, R Bartsch, Proc 3rd Symp Physics & Techn of Compact Toroids, Los Alamos NM, LA-8700-C (1980)
2. M Tuszewski, Nucl Fusion, 28, 2033 (1988).
3. R Milroy, J Brackbill, Phys Fluids 25, 775 (1982).

Presenting Author: T Intrator

[9] R Siemon, R Bartsch, Proc 3rd Symp Physics & Techn of Compact Toroids, Los Alamos NM, **LA-8700-C** (1980)
[10] M Tuszewski, Nucl Fusion, **28**, 2033 (1988).
[11] R Milroy, J Brackbill, Phys Fluids **25**, 775 (1982).

WeP12

Study of Physics of Compact Toroids in the Proposed SPIRIT Program

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The SPIRIT (Self-organized Plasma with Induction, Reconnection and Injection Techniques) device [1] has been proposed to explore the physics of CT's (Compact Toroids), especially FRC's (field reversed configurations) and low-aspect-ratio RFP's (reversed field pinches). The attractive features of this program are: (1) innovative formation schemes of CT's by merging of spheromaks; (2) flexibility in controlling shape, flows, ion kinetic effects, and boundary conditions for stability study; (3) CT sustainment by a combination of NBI and an ohmic transformer; and (4) ability to study boundary areas between different CT plasmas to find the optimal confinement concept. Study of FRC plasmas. One of the attractive features of the SPIRIT project is its ability to form FRC's with large flux (up to 50 mWb) on a time scale much longer than the Alfven time [2]. Stability of FRC plasmas can be studied with a wide range of elongation (0.5-4), s* (4-60), and flow (up to the Alfven speed). Simple rigid-body models [3] and extensive 3D hybrid or MHD simulations [4] are used to examine effects of elongation, (shear) flows, two-fluid physics, finite Larmor radius of bulk ion and energetic (beam) ions, and conductors are being examined for both global (tilt and shift) and interchange modes. One configuration has been found [4] to be stable to all n=1 modes when $0.5<E<1$ with a conducting shell. The higher n interchange modes are being examined by including ion orbits, profile effects and kinetic ion beams. A proposed scheme to sustain FRC plasmas in SPIRIT is by neutral beam injection (NBI). It is found that all 50-keV ions can be confined with reasonable parameters [5]. Thermoelectric effects due to an electron temperature gradient are found [6] to be favorable in sustaining the FRC configuration against resistive diffusion when central electrons are heated. A small toroidal field and 3D modulations can be introduced to study transition towards ST and compact stellarator concepts. Study of low-aspect-ratio RFP plasmas. A growing consensus in the RFP community is that suppression of magnetic fluctuations and stochasticity will lead to significant improvement in confinement. Consistent with the single helicity state found in RFX, one expect smaller number of resonant modes in the low-aspect-ratio limit. The SPIRIT device can test such a concept by introducing small TF coils through the geometric axis. Since the plasma will be formed by co-helicity merging of spheromaks, a much smaller OH coil will be sufficient to sustain RFP plasmas.

[1] M. Yamada et al., Bull. Am. Phys. Soc. 43, 1768 (1998).
[2] Y. Ono et al., Proc. of 1992 IAEA Meeting 2, 619 (1993).
[3] H. Ji et al., Phys. Plasmas 5, 3685(1998).
[4] E. Belova et al., invited talk at APS 1999 meeting at Seattle.
[5] D. Mikkelsen et al., Bull. Am. Phys. Soc. 43, 1768 (1998).
[6] A. Hassam et al., Phys. Rev. Lett. 83, 2969(1999).

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WeP13

Design and Fabrication of a High Density Field Reversed Configuration

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The design of the Field Reversed Configuration (FRC) experiment at Los Alamos National Laboratory takes advantage of our many years of experience with these devices. The physics of attaining a high density and temperature FRC requires magnetic energy that scales with the size of the Green Newton magnetic field. This in turn requires large bias magnetic field (<0.4 Tesla), large crowbar field (< 5 Tesla), and large theta pinch electric field (< 0.8 kV/cm). The FRC formation region is inside the theta pinch coil, and eventually the FRC will be translated along a guide field into a mirror trapped region inside the liner. The theta pinch coil is a single turn clamshell design with inside diameter approximately 12 cm. Larger bore symmetric cusp coils at each end and guide field and mirrror field solenoids for the translation region are designed as modular combinations of a standard magnet pancake unit. The initial design calls for slow but pulsed wave forms for all magnets so they can ring using capacitor banks. Physics and engineering requirements, and drawings along with a progress report will be presented.

Presenting Author: J.M. Taccetti

WeP14

Azimuthal Dynamics of Strong Ion Rings

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We present observations of proton rings containing up to 100 kA of 0.7 MeV protons produced by the FIREX experiment at Cornell. When the rings are produced with minimal axial velocity and maximum diamagnetism of about 10% of the applied solenoidal field, complex dynamics is observed which results, within about three cyclotron periods of the ring protons, in an reproducibly organized nonaxisymmetricspiral structure. This structure and its evolution are observed with an extensive array of magnetically insulated Faraday cup diagnostics. The observations will be shown and possible mechanisms discussed. We suspect this evolution is due to collective interaction of the proton current with alfven modes of the background plasma, and may be evidence of a strong instability in the azimuthal dynamics of the ring-plasma system.

Presenting Author: J.B. Greenly

WeP15

Detailed Modeling of Proposed Liner-On-Plasma Fusion Experiments

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Magnetized Target Fusion (MTF) is a potentially inexpensive approach to controlled fusion in which a preheated and magnetized target plasma is hydrodynamically compressed by an imploding liner. If electron thermal conduction losses are magnetically suppressed, relatively slow O(1 cm/microsecond) "liner-on-plasma" compressions, magnetically driven using inexpensive electrical pulsed power, may be practical. Target plasmas in the range 10^{18} cm^{-3}, 100 eV, 100 kG need to remain relatively free of potentially cooling contaminants during formation and compression. Magnetohydrodynamic (MHD) calculations including detailed effects of radiation, heat conduction, and resistive field diffusion have been used to model separate static target plasma (Russian MAGO, Z-pinch, Field Reversed Configuration) and liner implosion (without plasma fill) experiments. Using several different codes, liner-on-plasma compression experiments are now being modeled in one and two dimensions to investigate important issues for the design of proposed liner-on-plasma MTF experiments. The competing processes of implosion, heating, mixing, and cooling will determine the potential for such liner-on-plasma experiments to achieve fusion conditions.

LA-UR-00-124

Presenting Author: R.C. Kirkpatrick

WeP16

Scaling Laws For Fomation And Adiabatic Compression Of A High-Density FRC In The MTF Setting

D.D. Ryutov

Most of the experiments with field-reversed configurations (FRC) have been carried out with plasmas in the density range of ~ 10^{15} cm^{-3}, plasma radius ~ 10 cm, and the magnetic field strength ~3-5 kG. This set of numbers determines the other parameters of the experiment, e.g., the field-reversal time, the currents in the coils, and the voltages needed to drive these currents (and, therefore, the requirements to the insulation). It also specifies the value of the parameter s (equal, roughly speaking, to the ratio of the pinch radius to the ion gyroradius). We derive scaling relations that allow one to evaluate characteristics of the experiment in the case of higher densities and magnetic fields, and lower initial dimensions, typical for the MTF setting [1,2]. In particular, issues of plasma ionization, retention of the magnetic flux during the field-reversal stage, and radiative losses are considered.

We evaluate also the changes of dimensionless parameters governing plasma confinement, and discuss the evolution of these parameters in the course of adiabatic compression of the initial configuration by imploding liner. Three-dimensional implosions, maintaining the geometrical similarity of the plasma configuration, are considered. We show that, generally speaking, confinement characteristics improve in the course of the implosion.

The third group of scalings relates to the problem of the Rayleigh-Taylor instability of the plasma-liner interface, which becomes important during the stage where the plasma pressure begins to decelerate the liner. We discuss a possibility of eliminating this problem by operating the system in a mode where the desired plasma parameters would be reached before the deceleration begins. We evaluate the energy penalty related to this stabilization technique and find the scaling of the energy efficiency vs. the stability margin.

Work performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract W-7405-ENG-48.

1. R.P. Drake, J.H. Hammer, C.W. Hartman, L.J. Perkins, D.D. Ryutov. "Submegajoule liner implosion of a closed field line configuration." Fusion Technology, v. 30, 310 (1996).
2. R.E. Siemon, I.R. Lindemuth, K.F. Schoenberg. "Why Magnetized Target Fusion offers a low-cost development path for fusion energy." Comments on Plasma Phys. and Contr. Fus. v. 18, 363 (1999).

WeP17

The Recirculator Project at LLNL*

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The Heavy Ion Fusion Group at Lawrence Livermore National Laboratory has for several years been developing the world's first circular induction accelerator designed for space charge dominated ion beams. The accelerator currently extends ten half-lattice periods (HLP) with induction cores for acceleration placed on every other HLP. A network of Capacitive Beam Probes (C-probes) has also been enabled for beam position monitoring throughout the bend section. Experiments on the quarter ring have been completed with a successful demonstration of coordinated bending and acceleration. Apon completion of the ring, the Recirculator would offer a unique capability to the study of highly non-neutral plasmas over long period of times, approximately 300μs. In particular, the longitudinal confinement and manipulation of these plasmas is of primary interest. Data from experiments performed on the 90° bend section will be presented and possible experiments for the full ring will be discussed.

* This work has been performed under the auspices of the US DOE by LLNL under contract W-7405-ENG-48

Presenting Author: L.E. Ahle

WeP18

Planning for a Heavy Ion IRE

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The next major step in the U.S. inertial fusion energy program, the Integrated Research Experiment (IRE) will play a critical role in the development of inertial fusion energy. The IRE, together with the target results from the National Ignition Facility, must give sufficient confidence to proceed to the next step, which is to design and build an experimental test facility, demonstrating fusion power production. The IRE conceptual design effort is scheduled to begin in about two years, but preliminary design examples have already been made to act as computational models to develop simulation tools, and to explore possible high energy density experiments. We will review the algorithms which lead to the designs and give examples of parameters and capabilities of the IRE, as well as the questions that are being addressed in the near term research program that need to be resolved before a final design is achieved.

Presenting Author: John J. Barnard

WeP19

A Mini-Final Focus system for Inertial Fusion Energy

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One of the alternative scenarios for a heavy ion driver for inertial confinement fusion is final trasport as a self-magnetic pinch. A beam is stripped by passing it through a thin foil (or gas) which also supplies neutralizing confinement of neutralizing electrons. Immediately before the stripping foil a mini final focusing system would be applied, which makes the envelope waist of a few millimetre radius at the foil. This mini-focus could be applied at a substantial distance from the fusion chamber, prior to final compression. In this scheme, the usual chromatic aberration constraint on the heavy ion fusion driver can be relaxed by at least an order of magnitude, which gives substantial relief on the longitudinal beam dynamics issues. Transverse focal constraints are also greatly eased.

Presenting Author: E Lee

WeP20

Beam Splitting in a Heavy Ion IRE and Driver*

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The difficulty of focusing intense heavy ion beam onto a small target is determined by the temperature and the amount of space charge, as measured by the beam perveance, a quantity which is proportional to the ratio of the line charge density to the beam kinetic energy. Therefore the final focus of a heavy ion driver limits the allowable perveance per beam. This can push the accelerator design away from the cost-optimal beam perveance for transport and acceleration of beams. The ability to split the beams at the end of the accelerator could free the accelerator design from the constraints of the final focus and thus lower cost. But splitting must be done without too much beam heating, or focusability will be lost. In the planned heavy ion IRE (Integrated Research Experiments), beam energy will be significantly lower than in the driver, making the focus harder than it need be to show feasibility for a driver, and beam splitting would not only alleviate this, but would also enable better symmetry of beams at the target. This paper examines transverse 4-way splitting of beams. We will describe a proposed system, examine engineering limitations, and display PIC simulations of the change in the beam distribution function due to the splitting.

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Presenting Author: C.M. Celata

WeP21

Diode-Pumped Solid-State Lasers for Inertial Fusion Energy

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We have begun building the "Mercury" laser system as the first in a series of next generation diode-pumped solid-state lasers for inertial fusion research. Mercury will integrate three key technologies: diodes, crystals, and gas cooling, within a unique laser architecture that is scalable to kilojoule and megajoule energy levels for fusion energy applications. The primary near-term performance goals include 10% electrical efficiencies at 10 Hz and 100J with a 2-10 ns pulse length at 1.047 μm wavelength.

The Mercury laser amplifier head and gas cooled architecture has been designed in a modular and scalable fashion, with the laser slabs mounted in aerodynamic vane elements. Gas flows over the faces of the laser slabs in the cooling channel to remove the waste heat generated during the lasing process. The assembled slab and vane cassette are then inserted into the amplifier head. The first laser head assembly was fabricated and installed in the Mercury laboratory. In order to fully test the amplifier head and pump delivery system this year, surrogate gain media (Nd:glass) were placed in the vane elements within the amplifier head. This allows us to test the pump delivery efficiency, pump light uniformity within the laser slabs, gas flow dynamics, and thermal deposition profiles. Once the Yb:S-FAP crystals are ready, we can easily switch the surrogate slabs with the crystals. We have assembled one out of four pump delivery arms.

A critical technology for realizing inertial fusion energy is in the cost and efficiency of laser diode arrays. Existing diode technical performance specifications do not currently meet the demanding requirements of IFE. In addition, the manufacturing costs will have to be reduced by approximately two orders of magnitude to make IFE economically viable. Together with an industrial partner, Coherent-Tutcore, we made significant progress on the development of aluminum-free 900 nm laser diode bars. We packaged, characterized and life-tested arrays of 900nm laser bars using this diode material.

Significant progress has been made in understanding the growth characteristics and defect chemistry of Yb:S-FAP [$Yb^{3+}:Sr_5(PO_4)_3F$] crystals. The growth of full size crystals has been a challenge due to a number of defects, including: cloudiness in as-grown boules, bubble core defects, grain boundaries, and cracking in larger diameter boules > 4.0 cm. Results have produced sub-scale boules with greatly reduced defects that have optical properties that nearly meet the Mercury specifications.

A simple scaling of beam smoothing has been established which is helpful in the analysis of prospective ICF laser drivers. By applying spectral shaping to the amplifier input and broad band frequency conversion with dual triplers, the gain narrowed bandwidth can be increased to ~1 THz for DPSSLs. Such a laser amplifier (e.g. Mercury and its successors) is therefore promising for use as a driver in an IFE power plant.

The reliability, availability and maintainability of the laser components should be deemed to be acceptable for a future *integrated research experiment* or IRE (kJ-class laser coupled with average-power target chamber), and have a plausible means of attaining the driver requirements of inertial fusion energy (IFE).

Presenting Author: S.A. Payne

WeP22

A New Injector Concept for HIF Induction Linacs

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The two most prominent types of driver for inertial fusion energy are laser and heavy ion beams. Fundamentally, these two drivers require different architectures. Heavy ions must be accelerated to a few GeV for target penetration and to minimize the space-charge effect whereas high photon intensity can be achieved by adding parallel laser beams. Therefore the heavy ion beam approach is inherently not as modular as that of the laser. Furthermore, the cost per unit length of an ion accelerator is much higher at the low energy end, especially in the injector section. Thus it is necessary to find innovative ways to reduce component costs in order to create an affordable development path for heavy ion fusion (HIF).

For HIF induction linac drivers, the typical injector parameters are total beam current of 50-100 A, comprised of many (N approx. 50-100) individual beams of 0.5 - 1.0 A each, kinetic energy of 2 MeV, 10-20 micro-seconds pulse length, and a repetition rate of 10 Hz. An important aspect in designing the injector is dealing with the space-charge limits in beam extraction and in the subsequent low energy transport section. The traditional approach is to start with low current density (large radius) beams and gradually compress and steer them into the transport system. The major disadvantage of this approach is in the very large size, and likely very high cost, of the N-beam system.

Motivated by the success of neutral beam injectors for MFE, a new approach is now being investigated. The idea is to merge high current density miniature beamlets into the multiple 0.5 - 1.0 A beams before injection into the matching section. According to several scaling rules, the multiple beamlet approach is found to be more compact, although the requirements on the ion source are more demanding. Simulation study has shown that when the beamlet current is small and the number of beamlets is large, the emittance growth due to beamlet merging can be acceptable. By aiming and steering the beamlets to quickly match into the electrostatic quadrupole (ESQ) focusing channels, the matching section length and size can be greatly reduced. We compare injector designs based on these two methods and identify critical areas in developing components.

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Presenting Author: J. W. Kwan

WeP23

Advanced IFE Power Plants Utilizing Fast-Ignited, Tritium-Lean Targets*

Latkowski, J.F.

While traditional inertial fusion energy (IFE) target designs call for equimolar quantities of deuterium and tritium, some advanced, high-density ($\rho r \sim 10$ g/cm^2) target designs have been proposed with tritium inventories as low as 0.5%. To be viable at reasonable driver energies (~ 5 MJ), such targets would need to rely upon fast ignition techniques. At such high densities, a significant fraction of the yield is provided by D-D and D-^{3}He reactions, the neutron spectrum is softened, and the targets may be self-sufficient from a tritium breeding perspective. These features provide much more flexibility for innovation in power plant design.

Due to the self-sufficiency of the target and the low tritium inventories, a power plant utilizing such targets may have exceptional safety and environmental characteristics. This was demonstrated in a preliminary assessment for a particular set of design assumption [1]. Given that breeding materials may not be necessary, blanket materials may be selected by design criteria other than the tritium breeding ratio. Items such as vapor pressure (affects beam propagation), chemical compatibility, power conversion (higher temperature operation and/or direct conversion), pumping power (important for liquid wall chambers), and safety and environmental aspects can be emphasized in the selection of materials and chamber design. Significant additional work is needed to explore the design space of possible material choices, chamber configurations, and power conversion schemes for power plants using these targets.

[1] J. F. Latkowski, B. G. Logan, L. J. Perkins, W. R. Meier, R. W. Moir, S. Atzeni, and J. Sanz, "Safety and Environmental Advantages of Using Tritium-Lean Targets for Inertial Fusion Energy," Presented at the First International Conference on Inertial Fusion Sciences and Applications, Bordeaux, France (September 1999).

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Presenting Author: J.F. Latkowski

WeP24

Higher Vacuum Operation of a Liquid-walled IFE Chamber

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Recent calculations of stripping rates of heavy ions indicate that the mean-free-path, in the standard HYLIFE-II with a vapor density of $5x10^{13}cm^{-3}$ [1], may be reduced to about 50 cm [2] compared with the 120 cm used previously [3]. Although unconfirmed by experimental data at low ionization states, these results increase the incentive to reduce the vapor density along beam trajectories. Reducing the vapor density in a liquid-walled inertial fusion chamber, also increases the mean-free-path of neutralizing electrons co-moving with the beam (a superior method of neutralizing [4]), and reducing the plasma density within the chamber mitigates any possible instabilities.

We introduce a new concept of reducing the vapor density along the heavy-ion beam trajectories by lining the interior target region and separating columns of heavy-ion beams with dense "curtains" of cooled Flibe droplets. These curtains are in addition to the previous HYLIFE-II concept of sprays of droplets, external to the oscillating jets, which contribute to a faster pumpdown after each shot and allow a higher rep-rate [1]. The vapor density is reduced by exploiting both the ability of Flibe liquid to sorption-pump its vapor with unity sticking coefficient and the dependence of Flibe vapor pressure on the liquid temperature and composition (the vapor density is of $5x10^{13}cm^{-3}$ at 650 C, dropping by a factor of 2 every 25 C, resulting in a factor of 680 reduction in vapor pressure at the freezing temperature of 460 C). The cooler droplet curtains hide the 650 C oscillating jets from the target region, and form ducts either side of the crossed beam jets (~625 C) that reduce the vapor density on the scale length of the curtain separation.

We have calculated the steady-state vapor density distributions along the heavy-ion beam trajectories, for a variety of modifications to the HYLIFE-II configuation. Reductions in vapor line density by factors of several appear feasible with minor changes, factors of 10-50 may be feasible by reducing the mass and temperature of the beam jets. We find optimum temperatures for the Flibe droplets and beam-jets, based on feeding only those jets with cooled Flibe that has passed through heat exchangers. We are in the process of re-evaluating the time dependent pumpdown: the first millisecond is computed with the TSUNAMI code with the addition of ionization [5] and radiation transfer processes. The subsequent pumpdown is followed by a power and particle balance for the Flibe vapor and liquid, with a range of assumptions for the heat transfer into the liquid.

*This work performed under the auspices of the U.S. DOE by LLNL under Contract W-7405-ENG-48 and by LBNL under Contract No. DE-AC03-76SF00098.

1. R. W. Moir, et al., Fusion Technology 21, 1475 (1992); 26, 1169 (1994).
2. R. E. Olson, to be presented at HIF Internat'l Symp., San Diego, March 13-17, 2000.
3. D. A. Callahan, Fusion Engin. and Design 32-33, 441 (1996).
4. B.G. Logan, D. A. Callahan, Nucl. Instr. and Meth. in Phys. Res A 415, 468 (1998).
5. C. Jantzen, E. P. Lee, P. F. Peterson, "Parametric Studies of Ablation and Venting from the Central Cavity of the HYLIFE-II Reactor," Fusion Technology 34, 1047 (1998).

WeP25

Plasma-Channel-Based Reactor and Final Transport

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As part of an on going exploration of final transport schemes based on plasma channels, a point design of a final focus and reactor system is being developed. Six MJ of 3-4 GeV Pb ion are delivered to a target in two opposing current-carrying plasma channels within a modified HYLIFE II reactor filled with 5 Torr of Xe gas. The transition from the superconducting quadrupoles to the small entrance hole in the reactor is provided by a tapered plasma lens. The required channel parameters of 50 kA current and 10 mm size have been demonstrated to be feasible in recent Lawrence Berkeley National Laboratory (LBNL) experiments and detailed characterization of the channel dynamics is ongoing. The proposed system relaxes requirements on the incoming beam and provides life of the plant protection of all critical components around the reactor Engineering design work is paralleled with beam and channel simulations from the final quadrupoles to the target.

Presenting Author: S. Yu

Helical RFPS: Theory and Experiment

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It was previously shown that a pinch plasma helically wrapped tightly around a guiding axis acquires a large axial (toroidal) transform of magnetic line direction that reverses the pitch of magnetic lines near the plasma edge, without the need of azimuthal (poloidal) plasma current [1–3]. The transform arises from the combination of a strongly helical magnetic axis and a strongly "D"–shaped poloidal plasma cross section. This configuration was called the "Helical–D Pinch" [2,3]. It was conjectured that the replacement of poloidal plasma current, which in conventional reversed field pinches (RFPs) is driven by current transported outward from the central core by overlapping magnetic islands, will reduce MHD instabilities while maintaining the favorable high magnetic shear of the RFP, thereby improving RFP confinement.

Finn *et al.*, demonstrated the theoretical existence of steady–state, resistive, helical RFP equilibria in a *circular* conducting boundary with only toroidal loop voltage drive [4]. The equilibrium is the saturated state of a single, near–resonant, internal, ideal MHD mode. They identify "stellarator transform" as the origin of magnetic line pitch reversal near the plasma edge. The three–D resistive MHD computations yielded weakly fluctuating equilibria with large regions of intact magnetic surfaces. Therefore, confinement might be expected to be good. The strongly helical magnetic axis, the sign of its pitch and the substantial D–shape of internal magnetic surfaces (Fig. 8(c) of [4]) all match the helical–D prescription.

Quasi single-helical-mode states have recently been observed in the RFP experiment RFX [5]. Magnetic fluctuations are lower and the central temperature higher than in the conventional RFP phase of the discharge. Again, the helical magnetic axis, sign of its pitch and D–ness of internal magnetic surfaces match the helical–D.

In view of these and other recent theoretical and experimental results, it appears that the helical–D pinch has already been demonstrated serendipitously, albeit and sub optimally, in circular boundary systems. The helical axis and D–shape provide edge pitch reversal, and the numerical and experimental outcome is lower magnetic fluctuations and better confinement than conventional RFPs. A helical–D/RFP designed with an optimally shaped boundary ought to perform much better still. Ideas for such an experimental device will be presented.

[1] M.J Schaffer, U.S. Patent 4,734,247 (1988).
[2] M.J. Schaffer, IAEA Technical Committee Meeting on Innovative Approaches to Fusion Energy (Pleasanton, CA, USA, 1997 Oct 20–23).
[3] M.J. Schaffer, Comments Plasma Phys. Control. Fusion 18 (1999) 349.
[4] J.M. Finn, R. Nebel, C. Bathke, Phys. Fluids B 4 (1992) 1262.
[5] R. Pasqualotto *et al.*, 26th EPS Conf. (Maastricht, 1999) ECA 23J (1999) 1153.

Presenting Author: M.J. Schaffer

ThA2

Generation and Evolution of Plasma Flow in the MST Reversed Field Pinch

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Understanding the generation and evolution of plasma flow is an important part of understanding and improving many magnetic configurations. In the RFP, magnetic fluctuations play a major role in determining both the global magnetic configuration and the momentum profile. Several important mechanisms are emerging as important players. We have reported previously that resonant field errors exert torques on resonant modes. This is an important mechanism for plasma braking as in other devices. Equally significant, it appears, are the nonlinear torques operating between coupled modes. These play an important role in determining the flow profile and the time behavior of the flow at sawtooth crashes. Specifically, we find that coupling of two m=1 modes (resonant in the core) through an m=0 mode (resonant in the edge) results in a strong coupling of core and edge velocities when amplitudes are large. We have also begun measurements of the total magnetic component of the Reynold's Stress in the edge. Initial indications are that it is larger during sawtooth events than needed to explain the changes in plasma momentum. Indirect measurements indicate the primary component is due to locally resonant modes.

The response of the plasma to sudden changes in flow is also being investigated using biased probes. Strong edge flows are produced when the probes are biased and the response of the core flow is observed to follow on a slower timescale. The inferred plasma viscosity is much larger than classical estimates and is relatively insensitive to plasma density. The response of the core is slower during periods of low magnetic fluctuations implying that the coupling of the edge and core is mediated by magnetic fluctuations.

Several new diagnostics are being implemented on MST to measure flows with greater spatial resolution in both the core and edge plasma. These include a novel spectroscopic probe, charge exchange recombination spectroscopy, and Rutherford scattering. The status of these diagnostics will be presented. In addition, we are planning to apply two simultaneous and independent rotating magnetic perturbations to exert control over two coupled modes. This may give us some control of core flows and allow further investigation of nonlinear torques.

Presenting Author: D.J.G. Craig

ThA3

Highlights of Improved Confinement and Future Plans for MST

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Suppression of MHD magnetic turbulence remains the highest priority issue toward improving RFP confinement. To this end, progress continues in developing current profile control in MST. For example, using pulsed inductive current drive (PPCD), additional refinement of the parallel electric field pulse waveform has led to longer periods of fluctuation suppression and MST'a largest electron temperature Te=840 eV. Also the temperature profile peaks in the core, indicative of reduced core transport. In lower current plasmas, poloidal beta *12% and the confinement exceeds the "constant beta RFP scaling" characterizing past RFP performance. The universal and global affect on the plasma is dramatically illustrated by density fluctuations measured with FIR interferometry which strongly correlate with the magnetic fluctuations and sharply decrease during PPCD.

Although successful in reducing the targeted core-resonant (m=1) turbulence, these experiments also indicate control of edge-resonant (m=0) turbulence is required to achieve maximum confinement. Relative to previous PPCD experiments, the improved results cited above are due at least in part to suppressed m=0 activity. Similar sensitivity of confinement to edge-resonant modes is observed in cases with electrostatic current drive in the edge which primarily affects the m=0 modes. When the electrostatically driven current is directed for greater MHD stability, the sawtooth frequency is decreased, and energy confinement is somewhat improved. Hence, a more general view of turbulence suppression in the RFP is emerging, where both core and edge turbulence are important and must be controlled.

Looking ahead, rf current drive is hoped to provide more precise current profile control. An antenna to test lower hybrid current drive (800 MHz) has been installed in MST and will be tested at ~50 kW within the next year. Electron Bernstein wave current drive also looks promising, with low power launching tests planned in the near term. Although intended for current sustainment (one of the RFP's other major issues) Oscillating Field Current Drive might provide a means for steady-state edge current drive and profile control. A staged approach to a full sustainment test of OFCD is planned to proceed over the next several years. Frequency agility in the oscillators, especially in the lower power first tests, will allow controlled radial penetration of the driven current. These major system additions, plus substantial new diagnostic capabilities, form the basis of the expanded MST research program to be carried out over the next several years.

Presenting Author: J.S. Sarff

ThB1

Reactor Relevant Belt Pinches

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W. Dorland
Q.P. Liu

Short inductive pulse belt pinch experiments have obtained MHD stable β ~50%. (A belt pinch is a very high elongation tokamak: in the experiments above, aspect ratio A~5 and elongation $\kappa \sim$ 5-10.) Here we examine the reactor prospects for steady state, high bootstrap fraction belt pinches. The MHD equilibrium code TOQ finds that ballooning stable β is roughly given by $\beta = C\,\kappa\,/A$ where C~1/4 to 1/3, for 1.4<A<6 and $2 < \kappa < 6$ (e.g., =50 % for A=3 and $\kappa = 5$) for >90% bootstrap fraction cases. Very high elongation ($\kappa \sim$ 5-6) together with moderate to high aspect ratio require some inboard indentation for equilibrium code convergence. Vertical stability requires tolerably close conducting shells (b/a ~ 1.3) for most cases. Technologically feasible feedback control of resistive wall vertical motion requires adequate wall conductivity, which can be provided by ~ 2 cm liquid lithium. Solid conducting shells of the requisite conductivity are problematic in a D-T reactor environment. Technological approaches for both plasma facing free surface liquid metal and channelled liquid metal are discussed, as well as liquid metal MHD effects on plasma stabilization. Beta normal increases quite slowly with κ (due to strongly increasing I/aB) so prospects for kink stability with a shell appear good, and will be discussed if time permits. Some indentation also produces enough stable average MHD curvature so that island growth from bootstrap effects can be eliminated or reduced by an order of magnitude. Comprehensive gyrokinetic micro-instability calculations in numerical belt pinch equilibria will also be given. Finally, the feasibility of obtaining indentation without linked inboard poloidal field coils by using modular, non-planar TF coils (as in a modular stellarator) to create an axi-symmetric equilibrium with negligible coil induced islands is also discussed.

Presenting Author: M. Kotschenreuther

ThB2

Initial Results on ET

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Discharge cleaning plasmas have been produced in ET for the last 14 months. Initially only 20% of the TF coils were available along with 3% (in weight) of the Ohmic coils. The full TF coil became operational in the summer of 1999. With the addition of vertical field coils, fully equilibrated Ohmic plasma became available in Nov, 1999. Since then we have achieved 0.5 second MHD limited plasmas at a toroidal field of 800 Gauss and a loop Voltage of 400 mV. This performance is a bit better than expected even at the design parameters. Recently we have tested gas puffing using hydrogen, helium and argon gas. Excellent results were achieved in this reduced parameter operation with only a partial OH transformer installed. Density confinement times of up to 50 msec have been deduced from hydrogen puffing experiments in L-mode with titanium coated walls. And finally, bifurcated poloidal rotation was also achieved with 20 A radial current drawn from an internal electron source at -500 V bias. Negative radial resistivity is observed across the L-to-H forward and reverse transition, (again the results are somewhat better than expected). Fluctuation properties of ET are under study with reflectometers and Langmuir probes. Transport physics analysis and future expectations will be presented.

Presenting Author: R.J. Taylor

ThB3

The Concept of Tokamaks with the Lithium Walls

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At present, there is a significant frustration about reactor prospectives of the tokamaks which were studied extensively since late of 60s. Leaving aside nuclear engineering aspects of the tokamak reactors, it is possible to conclude that none of the most fundamental plasma physics issues, such as energy extraction from the plasma, plasma refueling, helium exhaust, controlling the stability and the electric current, has yet been resolved for the tokamak-reactors even at the conceptual level.

Over past two decades, the concept of the tokamak-reactor became entirely associated with the divertor configurations. In fact, the divertor is in conflict with many basic requirements. Thus, instead of distributing the power deposition on the inner wall, the divertor concentrates it. Instead of increasing the edge temperature of the plasma in order to suppress the anomalous transport, the divertor requires just opposite. For the helium exhaust, the divertor concept allows the thermalization of the alpha-particles, what makes them indistinguishable from D and T ions of the plasma. Then, the divertor concept relies on some magic properties of the highly non-equilibrium scrape of layer, which would resolve the problem of the power and helium extraction from the plasma.

The existing approaches also failed to solve the problem of the central refueling for the reactor control and again relies on some inefficient high tech means for affecting the fusion rate and the plasma profiles.

From the pragmatic point of view, the now dominant, biased toward the divertor, concept of the tokamak-reactors played a grave role in shutting down TFTR, the major US fusion experiment, as well as in destruction in of all the US tokamaks with the circular cross-section (TEXT, PLT and, now, TFTR).

Being alternative the the divertor, the concept of the tokamaks with the lithium walls (LiW), which is intrinsically originated from the results of TFTR experiments with the Li pellets, has been triggered by new ideas from the APEX program. In Dec. 1998, it was understood, that the lithium represents the unique metal, which can be propelled at the necessary speed along the plasma facing walls and, thus, can potentially solve the problem of the energy extraction. In agreement with the TFTR results, the lithium (solid or liquid) can dramatically affect the plasma confinement and stability regime by low recycling wall conditions and by enabling conducting stabilizing wall at the plasma boundary. For the first time, with the LiW, the expected plasma density and temperature profiles could be in consistency with suppression of anomalous energy transport and with enhanced stability, both required for the commercially efficient fusion reactors.

The concept of LiW tokamaks initiates in a new way the solution of the problem of the central refueling for the reactor control. It reveals the remarkable properties of the magnetized Morozov's rings, which can serve as the transporters of the fuel as well as the potential means for the helium exhaust (see two separate abstracts for ICC2000 on Morozov's ring).

The the key points of the LiW tokamak concept will be presented to ICC2000.

Presenting Author: Leonid E. Zakharov

ThC1

Status of the LDX Project

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Columbia University
J. Kesner, P. Michael, J. Minervini, A. Radovinsky, J. Schultz, B. Smith, A. Zhukovsky
MIT
R. Ellis, J. Schmidt
PPPL.

The Levitated Dipole Experiment (LDX) is a joint Columbia University/MIT research project which will investigate for the first time in a laboratory device the possibility of steady-state, high-beta dipole confinement with near classical energy confinement. To carry out this research, a new experiment is under construction at the MIT PSFC, and all of the major components of the experimental facility are being or have already been fabricated. The experiment consists of a high-performance, Nb3Sn superconducting "floating ring" which will be levitated magnetically within a large vacuum vessel. The cryostat of floating ring should allow levitation for more than six hours. The cryrostat is made with three-concentric tori: a high-pressure helium vessel containing the Nb3Sn conductor, a radiation shield made from fiberglass and lead, and an outer vacuum vessel. The superconducting floating ring will be charged inductively with a second superconducting coil located at the bottom of the vacuum chamber. A mechanical lifting system is located along the axis of the vacuum system and will be used to position the dipole magnet for initial experiments and provide safety against control faults during ring levitation. The levitation coil will be a hybrid coil made from a small water-cooled copper coil and a high-Tc superconducting coil. Plasmas will be produced using multiple-frequency ECRH. The microwave heating will initially create, high-beta, hot-electron plasmas, and the multiple ECRH frequencies will be used to adjust the radial pressure profile. The experimental program is designed test the stability and confinement properties of this configuration. The initial plasma operations are expected to take place next fall. We will phase in experimental operations leading up to levitated coil operations as follows: (1) low current in the "floating" ring and operation with the ring mechanically supported, (2) high current (1.5 MA) operation with a supported ring, (3) high current operation with a levitated ring. The presentation will review the design, fabrication, and experimental plans for the LDX experimental device.

Presenting Author: D.T. Garnier

ThC2

Centrifugal Confinement for Fusion: The Maryland Centrifugal Torus (MCT)*

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The basic idea of centrifugal confinement is to use centrifugal forces from supersonic rotation to augment the usual magnetic confinement. Using this extra "knob" optimally results in a device that features four advantages over tokamaks: steady state, no disruptions, superior crossfield confinement, and a simpler coil configuration. One geometry, to be used in the proposed MCT expt at the University of Maryland, is a simple mirror with a central column. The central column is biased with respect to the outer wall resulting in a radial electric field that drives toroidal rotation. The radial centrifugal force keeps the ions completely confined against parallel effusion, bead-on-wire fashion. The electrons are held in electrostatically but the potential well is much deeper than in a mirror machine, scaling as the square of the Mach number.

A second prong of the centrifugal idea is that the large velocity shear will stabilize even the flute interchanges. Thus, extra magnetic fields may not be necessary, although numerical simulations of flute interchanges indicate a residual wobble. If necessary, the latter could be suppressed by a weak toroidal magnetic field. The velocity shear will also quell microturbulence, leading to fully classical confinement as there are no neoclassical effects. The parallel electron transport is minimized by a large Pastukav factor resulting from the deep potential well. At Mach 4-5, the Lawson Criterion should be accessible.

The central goal of the MCT experiment will be to obtain MHD stability from velocity shear. Specifically, it will be determined how much if any toroidal field is necessary to suppress residual wobbles and convection from the interchange. Previous experiments were probably MHD convection limited and did not have a toroidal field. In addition, the MCT experiment will feature a plasma of elongation 6-8, which should reduce the interchange growth rate and so reduce Mach number requirements. The experiment will be small-scale (plasma width ~ 10 cm), possibly neutral dominated, but sufficiently in the MHD regime to test the above. An elongated mirror geometry using existing coils and power supplies cost-effectively is planned. A toroidal field capability will be included. Mirror ratios of 4-10 are planned with a maximum field of 2T. Voltage drops of 10-20 keV will be used. Zero-D transport code estimates suggest an operating regime where the various MHD issues above could be adequately tested. In addition, the experiment will study and optimize plasma formation and insulator design.

*Work supported by DOE

Presenting Author: R.F. Ellis

ThD1

Plasma Formation Studies and Plans for the Pegasus Toroidal Experiment

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The PEGASUS Toroidal Experiment is designed to explore techniques to minimize the center column of a spherical torus while maintaining good stability and confinement at very high beta. Initial studies are focused on stability boundaries as $A \rightarrow 1$ at both high and low toroidal beta, with an emphasis on edge-q and kink stability limits as functions of plasma geometry. Achievement of high I_N and high TF utilization as $A \rightarrow 1$ will be the focus of these efforts followed by the exploration of high β_t stability limits. Studies at relatively high TF and modest I_p/I_{tf} should allow access to ballooning limits in the range of $\beta_N \geq 6$, while operation at lower field may access near-unity β_t and regions of high I_N where wall stabilization is not required for external stability. The PEGASUS Toroidal Experiment is uniquely poised to explore the tokamak/spheromak transition regime in the near future. To this end, a new low-inductance toroidal field coil set will allow transient exploration of the $I_p/I_{tf} > 3$ regime and associated plasma relaxation phenomena. Initial operations are focused on startup plasmas and discharge evolution control, where $I_p \sim 0.1$ MA has been achieved with $I_p/I_{tf} \sim 1$, $A = 1.15 - 1.4$, $R = 0.25 - 0.35$ m, and at $B_t = 0.07$ T. High current ramp rates are observed (30 - 200 MA/s) with correspondingly highly elongated plasmas (> 3) both at high (0.07 T) and low (0.04 T) toroidal field. At ramp rates ≥ 30 MA/sec, a large-scale MHD instability, identified as a double tearing mode on the predecessor MEDUSA experiment, occurs during the formation stage and limits the ultimate current achieved. Many plasmas terminate with a series of Internal Reconnection Events (IRE's). Magnetic reconstruction is accomplished using TokaMac, a plasma equilibrium reconstruction code, which incorporates measurements from a Rogowski loop, magnetic pickup coils, and flux loops. Time-evolving currents in the vacuum vessel wall are modeled as a set of mutually coupled axisymmetric current filaments. Initial magnetic reconstructions indicate β_t on the order of 15% have been achieved. Completion of the power systems is in progress to allow operation at full pulse length (0.04 s) and plasma current (0.3 MA), to provide a target plasma for the Higher Harmonic Fast Wave (HHFW) heating system, and to provide access to high β regimes. Non-inductive startup and sustainment techinques are also being developed, including current injection via plasma guns and radio frequency heating and current drive using either Electron Bernstein Waves or HHFW.

Presenting Author: T.A. Thorson

ThD2

Overview of Experimental Results on NSTX

Masayuki Ono and The NSTX National Research Team
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The National Spherical Torus Experiment (NSTX) is a national fusion science facility whose mission is to establish the fusion physics principles of the innovative spherical torus (ST) concept. Physics outcome of the NSTX research program is relevant to near-term applications such as the Volume Neutron Source (VNS) and burning plasmas, and future applications such as the pilot and power plants. The NSTX facility is located at Princeton Plasma Physics Laboratory (PPPL) utilizing the TFTR (Tokamak Fusion Test Reactor) infrastructure. The NSTX facility is managed by PPPL and the National Research Team composed of researchers from over 14 institutions are carrying out the experiments on NSTX. The nominal NSTX plasma parameters are R0 = 85 cm, a = 67 cm, R/a * 1.26, BT = 3 kG, Ip = 1 MA, q95 = 14, elongation k * 2.2, triangularity d * 0.5, and plasma pulse length of up to 5 sec. The plasma heating / current drive (CD) tools are High Harmonic Fast Wave (HHFW) (6 MW, 5 sec), Neutral Beam Injection (NBI) (5 MW, 80 keV, 5 sec), and Coaxial Helicity Injection (CHI). The NSTX device began the plasma operations in February 1999 with the NSTX First Plasma Milestone successfully achieved on Feb. 15, 1999 ten weeks ahead of schedule. The NSTX Construction Project was successfully completed on the Budget (Total Project Cost) and excellent safety record. With the double swing 6 kV OH power supply, the plasma current was successfully ramped up to the device design value of 1 MA on Dec. 14, 1999 about 9 months ahead of schedule. Key to the achievement of high current plasma discharges was the implementation of the real time plasma control system in collaboration with GA. The Skybolt I computer system was able to feedback control on the plasma radial and vertical positions as well as the plasma current. This enabled the research team to study the ohmically heated ST plasmas up to 1 MA. The CHI experiments on NSTX has started in Nov. 1999. Plasma currents of up to 133 kA were produced using about 20 kA of injected current. Stable CHI discharges of up to 130 msec has been produced, a record for CHI. For the longer range, the injector current will be increased toward 50 kA level producing up to 500 kA of CHI discharges. The HHFW heating experiment has also started in Nov. 1999. Using eight antennas connected to two transmitters, up to the 2 MW of rf power was successfully coupled to the plasma. The ultra-soft x-ray diagnostic shows an indication of core electron heating during the HHFW modulation experiment. The system is designed to eventually deliver 6 MW using 12 antennas and 6 transmitters. The NSTX device at present (Jan. – June, 2000) is undergoing installation of Neutral Beam Injection system and Multipulse Thomson Scattering System. The NBI heating system and associated NBI based diagnostics such as the CHERS will be operational in the fall of year 2000.

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Presenting Author: M. Ono

ThD3

Coaxial Helicity Injection for the Generation of Non-Inductive Current in NSTX

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The Spherical Torus is an emerging plasma confinement concept that has the advantages of high beta, increased stability and a projected high fraction of bootstrap current drive. These favorable properties of the STs arise from its very small aspect ratio ($A \leq 1.5$). Elimination of the central solenoid is a serious consideration for future ST designs. This requires the demonstration of non-inductive plasma creation, and sustainment by non-inductive current drive schemes. Coaxial Helicity Injection (CHI) is a very promising candidate for initial plasma generation and for edge current drive during the sustained phase. The first experiments to explore this concept were successfully conducted on the HIT and HIT-II experiments at the University of Washington [1]. On NSTX, CHI involves driving current along poloidal flux that links the lower divertor plates. A 25kA, 1kV DC power supply was connected directly across the lower divertor plates to drive the injector current. On NSTX the central column and the inner divertor plates are insulated from the outer shell and the outer divertor plates, as on HIT. In the presence of a toroidal field, the J X B force resulting from current flowing along the poloidal flux can stretch the magnetic fluxes into the confinement chamber to create the desired magnetic configuration. An important goal for the NSTX program is to conduct the proof-of-principle experiments to validate this concept on a device that has a volume about 30 times that of HIT. Initial experiments on NSTX have successfully generated 130kA of CHI produced toroidal current using about 20kA of injector current verifying that there are no known limitations on extrapolating CHI to larger experiments. The best current multiplication factor thus far obtained has been 10. Stable discharges lasting for 0.13 seconds have been produced using pre-programmed coil currents and at vessel neutral densities compatible with high recycling divertor operation. Radiated power profiles indicate a hollow profile, implying that the bulk of the CHI driven current may be driven away from the center; this is an anticipated result. Different flux configurations for the generation of the seed plasma have been tried. MFIT (Magnetic Fitting code) [2] analysis of some of these early discharges indicates as much as 20% of the current along closed flux surfaces. Next step plans are for the production of optimized discharges at higher currents and lower densities to increase the closed flux fraction and to understand the related mechanisms from new diagnostics. The CHI produced plasma will then be driven using Neutral Beams and HHFW for heating and current drive, to demonstrate the concept of current generation and sustainment without the use of the Ohmic coil. Other experiments will focus on using CHI to add an edge current drive to an existing single null Ohmic discharge for the study of confinement and stability during the presence of an edge driven current.

[1] Jarboe T. R. et al., Phys. Plasmas 5, 1807 (1998).
[2] Lao L. et al, Nucl. Fusion 25, 1421 (1985).

Presenting Author: R. Raman

ThD4

TSC Modeling of NSTX

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The Tokamak Simulation Code [TSC] was used extensively in the design of the National Spherical Torus Experiment [NSTX]. Here we report on the results from detailed modeling of the first series of experiments on NSTX and show comparisons between code predictions and the experimental data. The agreement in the plasma current and the axisymmetric plasma motion between the data and the code can be quite good for reasonable assumptions regarding density profiles, impurity content, and energy confinement. Regions where the TSC predictions deviate from the experimental data indicate the presence of non-axisymmetric activity, and possibly the formation of runaway electrons at early time. We also present initial results of modeling the open field line current during the CHI experiments on NSTX, and on modeling future experiments involving non-inductive current drive.

Presenting Author: S.C. Jardin

ThP1

Physics Innovations in Spherical Torus Plasmas*

Martin Peng,
ORNL on assignment at PPPL

Recent theoretical and experimental studies have suggested that central plasma betas around 100% could be feasible in Spherical Torus (ST) plasmas. Such plasmas could be highly diamagnetic with a strong magnetic well near the magnetic axis, and/or could have extremely low aspect ratios (approaching 1.1) to resemble Field Reversed configuration (FRC) plasmas modified by a relatively small applied toroidal field. At high plasma temperatures and large sheared flows, the properties of this plasma becomes subject to the effects of comparable Alfven and sound speeds and a large dielectric constant. The plasma flow (V dot grad V) term is no longer neligible compared to the J x B and pressure gradient terms in MHD equilibrium. Large changes in MHD equilibrium solution are expected to affect the MHD and microinstabilities of the plasma, which in turn affect the plasma turbulence and transport properties. The large dielectric constant is expected to afford new features in rf wave launching, propagation, and absorption, such as for High Harmonic Fast Wave, Electron Bernstein Wave, and Ion Bernstein Wave. These properties are expected to be common to all high-beta Compact Toroid plasmas at high plasma temperatures. The implications of such properties on future power producing plasmas will also be discussed.

*Work supported by USDOE.

Presenting Author: Martin Peng

ThP2

Electron Bernstein Waves (EBW) in Overdense Plasmas as a New Tool For Advanced Tokamak and Spherical Torus Devices

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Electron Bernstein Waves (EBW) in Overdense Plasmas as a New Tool For Advanced Tokamak and Spherical Torus Devices* P. C. Efthimion, J. C. Hosea, R. Kaita, R. Majeski, C. K. Phillips, G. Taylor, J. R. Wilson, B. Jones, J. Menard, T. Munsat ---Princeton Plasma Physics Lab

The steady-state Advanced Tokamak (AT) and Spherical Torus (ST) require steady-state tools to control the current and pressure profiles. Electron cyclotron current drive (ECCD) and fast wave current drive are very inefficient in generating current. Lower hybrid current drive (LHCD) is more efficient, but is difficult to apply to the (ST) with $\omega_p >> \Omega_c$. There is also a need for additional external heating scenarios to be developed for the (ST). Other advanced confinement systems can benefit from new heating and current drive techniques. Electron Bernstein Waves (EBW) is emerging as a new heating and current drive tool. The EBW is an electrostatic wave that damps at the local electron cyclotron layer. The absorptivity of EBW is extremely high [1] because it is an electrostatic wave with a large k_{imag}. For example, NSTX will have an optical thickness $\tau > 3000$ and CDX-U will have $\tau > 300$. One can reach single pass absorption with a plasma density $> 10^{11}$ cm^{-3} and $T_e > 1$ eV. There are a number of means of accessing the EBW. Experiments on W-7Sdemonstrated EBW heating by O-X-EBW. The O-mode couples to the X-mode at a turning point when launched at a specific angle to the magnetic field. Then the X-mode efficiently mode converts to EBW at the upper hybrid layer. Recently, we demonstrated X-mode tunneling to the upper hybrid layer and mode converting to the EBW by measuring blackbody electron cyclotron emission in over dense plasmas on CDX-U. The tunneling efficiency can be high in CDX-U and other ST's when the edge density gradient is short because the tunneling length is only millimeters in length with $\omega_p >> \Omega_c$. This mechanism of accessing the EBW can be used by many magnetically confined plasma devices with $\omega_p >> \Omega_c$. More recently, Forrest, et al. [2] completed ray tracing and quasi-linear calculations suggesting EBW can bean effective current drive scheme. EBW has the potential to have the localization of ECCD, but with the efficiency of LHCD because it is an electrostatic wave. With these characteristics EBW it has the promise to become an effective tool for the advanced tokamak and spherical torus. With its low magnetic field the spherical torus requires low frequency sources for heating and current drive instead of state-of-the-art gyrotrons.

* Work support by US DoE contract No.DE-AC02-76CH03073.

[1] J. C. Hosea, et al., Phys. Rev.Lett. **39**, 408 (1977).
[2] C. Forrest, et al., submitted for publication to PoP.

Presenting Author: P.C. Efthimion

ThP3

A Spherical Torus Nuclear Fusion Reactor Space Propulsion Vehicle Concept for Fast Interplanetary Piloted and Robotic Missions

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An updated conceptual vehicle design of a spherical torus fusion reactor-based space propulsion system is proposed as a means of accomplishing a variety of interplanetary missions: from piloted multi-month trips between the outer planets to edge of the solar system robotic probes. The concept will emphasize design of major systems, including propulsion-driven modifications to the fusion reactor, power utilization (propulsion, auxiliary power usage, and waste heat rejection), primary structure, start-up method, thrust generation, propellant tankage, vehicle layout, performance, and mission design.

The design point of departure will be a small aspect ratio fusion reactor, envisioned in the year 2035+ time frame. In order to reduce mass, modifications to the reactor include removal of vacuum structure and some radiation shielding. Alternate approaches to reduce power conversion system size and mass from last year's concept will be investigated. Other potential modifications from last year's design include incorporation of bootstrap current overdrive operation, high harmonic fast wave plasma heating, technologies of lower system mass, and magnetic nozzle reaction control. Vehicle concept will include ignited (steady state) operation, refinement of the reactor dimensions and power balance characteristics by utilizing advanced fuels together with profile averaging of plasma temperature, number density, and current density. Future high temperature ceramic superconductors, anticipated to coexist with fusion technology, will be examined for their potential to serve as primary magnets. Recently initiated analysis and proof-of-concept experiments utilizing OSU's 1.8 MJ capacitor bank/magnetic nozzle test facility will be discussed. Supporting magnetic nozzle theory development, including anomalous resistivity effecting plasma-field line boundary layer behavior, was recently initiated at LANL and will be discussed. The potential use of Coaxial Helicity Ejection to extract reactor plasma power for thrust application will be discussed, as is the recently initiated plans to use the NSTX at PPPL for a limited proof-of-concept experiment.

Estimates of mass properties will include all primary systems: fusion reactor, primary structure, power processing, thrust generation, D3He fuel pellet injection, start up, tankage/insulation, reaction control, communications, avionics, and interface hardware. Estimates of power balance will include analysis of total power produced, quantity and form of radiation and charged power output, and power utilization (thrust, waste rejection, auxiliary, etc.)

NASA recently announced that a modest, dedicated fusion propulsion research program is likely to be initiated in FY2001.

Presenting Author: C.H. Williams

ThP4

Internal Field Measurements and Magnetic Reconstruction in SSPX

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In this paper we will discuss the measurement and MHD reconstruction of the internal magnetic field profiles in the SSPX spheromak. The field profiles can be used to derive the poloidal and toroidal current distribution in the spheromak, which can significantly impact the MHD stability of the device. The equilibrium of a sustained spheromak will be calculated based on edge field data and an internal field measurement from a new diagnostic called the Transient Internal Probe (TIP). TIP was developed and used on the Helicity Injected Torus at the Univ. of Washington. It consists of a magneto-optic probe sensitive to Faraday rotation that is sent through the plasma at a high speed. Laser light is passed through this probe and the resulting polarization measured. This gives a spatially resolved snap-shot of the field parallel to the direction of propagation. In SSPX, this diagnostic will be used to determine the toroidal field from the flux conserver to the magnetic axis. These data will be combined with data from an array of 14 edge magnetic probes mounted in poloidal and toroidal arrays to carry out the reconstruction.

CORSICA, a LLNL code designed to solve the Grad-Shafranov equation, will be used to fit these data to the field profile consistent with the geometry of the SSPX coaxial injector and flux conserver. At present, we use data from the edge probes only, along with manual iteration to arrive at the best fit for the magnetic field profile. We find that the field profile during the decay phase is in good agreement with the simple Bessel function model for the spheromak Taylor relaxed state. During buildup, the results are consistent with a flat $\lambda=\mu_0 j/B$ profile in the core, driven by an edge λ two times higher, as determined by the current and field in the coaxial source. We also find that the magnetic fields are fairly toroidally symmetric, even during buildup.

In addition, the amplitude of magnetic fluctuations at frequencies up to a few hundred kHz can be mapped out from the edge of the plasma to the magnetic axis. This will allow us to study how these modes, which are often seen in spheromaks using edge probes, may couple into the core of the plasma to maintain helicity balance and sustain the plasma current.

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ThP5

Particle Control Experiments in SSPX

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The Sustained Spheromak Physics Experiment (SSPX) addresses the physics of spheromak formation, current buildup, sustainment and decay. Particle control (of impurities and fuel gas) is essential for good spheromak performance since current buildup is a sensitive function of plasma resistivity $\eta \sim Z_{eff}\, T_e^{-1.5}$. The power balance between ohmic heating and radiative losses leads to a T_e requirement for the minimum normalized current density, $j/n > 10^{-14}$ A-m.

Spheromaks typically have close-fitting walls to provide MHD stability. This proximity can lead to enhanced plasma-surface interactions, for example, sputtering of hi-Z wall metals and desorption of loosely bound low Z impurity atoms. In SSPX, impurity control is obtained by tungsten spray coating of the walls, Ti gettering of the lower vessel, baking, glow-discharge cleaning, and attention to good vacuum practices.

Impurity line radiation measurements in the VUV show O, C, and N as the major contributors. Emission goes up with increased bank energy, but is insensitive to the amount of gas injected or prefilled for plasma discharge. Bolometric measurements indicate a large fraction (>50%) of input power is radiated away.

Density control in SSPX largely results from careful design of the flux conserver, magnetic field shaping, and the same surface conditioning used to reduce impurities. Open field lines intersecting the flux conserver are minimized by smoothly shaping the flux conserver to match magnetic fields produced by a main solenoid coil and two shaping coils. The flux conserver is 0.5 inch thick copper to reduce diffusion of field into the walls. Surface treatments (baking and GDC) reduce the wall gas inventory and the recycling coefficient.

At present, density during formation reaches as high as $2\text{x}10^{21}$ m^{-3}, which then falls rapidly to $\sim 2\text{x}10^{20}$ m^{-3} several 10s of microseconds after the injection current drops below the critical injection value and the spheromak detaches from the source. The density then stays relatively constant for 0.5 msec and then decays linearly to zero during the next 0.5 msec. In the detached phase, the density and density decay rate is insensitive to initial gas puff pressure or initial prefill conditions. Normalized current density j/n in the decay phase has been measured at $\sim 2\text{ x}10^{-15}$ A-m, confirming conditions are below that required for radiation burnthrough.

Plans are underway to further condition the plasma-facing wall by applying Ti gettering to the inside of the flux conserver. This has been shown to greatly reduce impurity radiation on SPHEX and other spheromaks. Studies will be performed of puffing additional gas during the discharge to address the possibility of gas starvation and wall scavenging. Increasing the pulse length is also planned to study buildup and sustainment of the spheromak. Scaling studies will be performed to see if j/n and T_e improve with the longer pulse length and lowered radiation.

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Presenting Author: H.S McLean

ThP6

Modeling of Spheromak Plasma Buildup in SSPX by Power Balance and Helicity Injection

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Results from spheromak plasma formation experiments in SSPX have been used to model plasma buildup during upcoming sustained plasma experiments. Presently achieved spheromak parameters, using only the formation bank, are I_{tor} ~0.6 MA, $W_{mag} \approx 16$ kJ, and $\tau_{mag} \approx 0.34$ ms. With operation of the sustainment capacitor bank, plasma duration up to ~3 ms will be possible. Gun voltage and power, calculated from SPICE code circuit calculations of capacitor bank output, were used to project sustained plasma parameters using complementary helicity balance and power balance models. These calculations were carried out for several gun magnet flux configurations. Control of plasma density and plasma impurities will be important for obtaining increased magnetic field decay times and significant buildup. Machine baking, as well as upgrades in glow discharge cleaning, and implementation of Ti plasma wall gettering are underway for improving wall conditions. Projections for ~100 eV electron temperature plasmas are $I_{tor} \geq 1$ MA and $W_{mag} \geq 60$ kJ.

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Presenting Author: B. Stallard

ThP7

Simulations and Modeling of SSPX Plasma Evolution

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We are undertaking three-dimensional, time-dependent nonlinear magnetohydrodynamic NIMROD simulations of spheromak plasma evolution using a plasma gun geometry[1]. Guided by promising looking relaxed-state axisymmetric equilibria obtained with the CORSICA code, we vary the boundary conditions (i.e., the distribution of frozen-in flux) and shape of the spheromak container in NIMROD simulations and determine their influence on the final state of the plasma. We hope to find better closed flux surfaces, thereby improving plasma and energy confinement. Thus, surface-of-section plots of the magnetic field lines are of particular interest as a diagnostic. The CORSICA code is also being used to study equilibrium and linear MHD stability properties of SSPX plasmas, and to analyse experimental data. We will report progress on various SSPX modeling activities.

We have also been examining fluctuation data from SSPX to find the phenomenology of the spheromak dynamo. Specifically, we have been analysing the signals from 8 equatorially positioned Rogowski loops surrounding the plasma. For methodology, we use the standard Fourier transform, and also a new Phase Velocity Transform technique [2]. The shots analysed have shown prominence of a very non-sinusoidal n=1 mode, a somewhat less strong n=3 mode, and a very much weaker n=2 mode. We suggest that the n=1 mode is related to a kink instability of the geometric axis, and the n=3 mode is an island resonant on a rational surface in the annular portion of the plasma. The absence of an n=2 mode suggests that the q profile lies above q=1/2 across the minor radius.

* Performed by LLNL for US DOE under Contract W-7405-ENG-48.

1. C.R: Sovinec, et al., Bull. Amer. Phys. Soc. 44, 112 (1999) and http://nimrod.saic.com/
2. N. Mattor, submitted to Phys. Rev. Lett., 1999.

Presenting Author: B.I. Cohen

ThP8

Steady Inductive Helicity Injection Current Drive

T. Jarboe

A new Steady Inductive Helicity Injection (SIHI) current drive method, which can be used to sustain spherical torii, spheromaks, and RFPs, is presented. SIHI has the following properties: a) helicity is injected at a nearly constant rate; b) neither magnetic energy nor helicity flow out of plasma at any time; c) no field lines penetrate the walls; d) the equilibrium is produced in a close-fitting flux conserver; e) a rotating n=1 magnetic structure for current drive is produced directly; and f) in the frame of the rotating field the current profile is nearly time independent and near optimum for a high beta spheromak and a spherical torus. SIHI can be applied to any toroidal plasma. The SIHI application to a high beta spheromak is a conservative step in the development of helicity injection current drive since this concept is an inductive version of the successful m=1 helicity injection experiment performed at the Los Alamos National Laboratory. The geometry is modified slightly to be compatible with a high-beta flux conserver shape because pressure-driven modes were observed to severely limit spheromak confinement. The design also has the capability of doing rotating field current drive. Recent results show that rotating field current drive may be the mechanism for closed flux current drive in HIT. Finally, Cowling's Theorem requires a non-axially symmetric motion of the magnetic axis to achieve current drive. Driving this motion directly instead of exciting an n=1 instability may be more energy efficient and less damaging to confinement.

Presenting Author: T. Jarboe

ThP9

Experimental Studies of Spheromaks at the Berkeley Compact Toroid Experiment

Edward C. Morse and Charles W. Hartman
University of California, Berkeley

The Berkeley Compact Toroid Experiment (BCTX) is a spheromak device with a 70 cm flux conserver and a Marshall gun plasma source. Radiofrequency (RF) heating in the lower hybrid range (432 MHz) is used to study the electron heat confinement using a 20 MW, 100 us power pulse. While electron temperatures to 200 eV have been measured with RF heating, the impact on magnetic decay from the RF heating is relatively weak. A simple model based upon parallel electron heat transport along stochastic magnetic field lines gives a scaling law for magnetic decay which fits the experimental data. Results of two-dimensional MHD code calculations are given which match many details of the experimentally observed formation physics. A novel breakdown enhancement scheme using Penning Ion Gauge (PIG) mode will also be described, which has been used successfully to lower the density and allow a mode of operation with lower impurity levels.

Presenting Author: E. C. Morse

ThP10

ET Magnetic Configuration and Equilibrium

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The UCLA Electric Tokamak is a low field (.25 T) device with large aspect ratio (A ~ 5). The machine is designed to operate with four sets of independent poloidal field coils to provide for OH current drive, vertical equilibrium field, plasma elongation (kappa < 2) and plasma shaping (D or reverse-D). The OH system should provide 10 V-s using a 10 KA power supply. A vertical field of up to .1 T is necessary to provide equilibrium in high beta plasmas. All the coils are located outside the vessel and are constructed from aluminum. The toroidal field coil is completed and is now operated at half field. Plasma shots of 1/2 second are obtained using a partial poloidal coil system as construction continues. Machine diagnostics include a Rogowski probe, loop Voltage monitors, Hall probes, visible and UV radiation detectors and CCD cameras. ET equilibria from codes and from initial plasmas will be presented.

Presenting Author: J.-L. Gauvreau

ThP11

Simulated Poloidal Rotation Effects on Kink Modes for the Electric Tokamak

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The Electric Tokamak (ET) will use poloidal rotation as part of its path towards towards a near unity beta omnigenous tokamak plasma. At UCLA, we are currently using a variety of tools within the theory community to explore the effects that this driven rotation will have on both turbulence and MHD stability. It was previously determined from gyrokinetic simulations that poloidal rotation faster than the poloidal unity Mach number can eliminate trapped-ion ITG modes likely to dominate our plasmas at that point [1]. We now turn our attention to beta limiting MHD instabilities such as global kink modes. We have used both NIMROD [2] and a reduced MHD version of FAR [3,4] to create a near global 1/1 resistive mode as a test mode ($S = 2 \times 10^5 - 1 \times 10^6$, and $r(q=1)/a = 0.6$) for a realistic ET aspect ratio and size. Externally imposed equilibrium poloidal rotation is then imposed on this mode. We find that the rotation profile shape does have significant effects on this test mode. Preliminary results suggest that rotation can *either* reduce the growth rate or enhance it depending on where the rotation maxima is relative to the q=1 surface. Other considerations of rotation such as effects on equilibria and other modes are also discussed in this context.

This work was funded by US DOE Grant no. DE-FG03-86ER53225, tasks I and II.

[1] M.W. Kissick, et al., Phys. Plasmas **6** (1999) 4722.
[2] A.H. Glasser, et al., Plasma Phys. Control. Fusion **41** (1999) A747.
[3] J.A. Holmes, et al., Phys. Fluids **25** (1982) 800.
[4] E.D. Held, et al., Phys. Plasmas **6** (1999) 837.

Presenting Author: M.W. Kissick

ThP12

Laser Ignition of an Isentropically Compressed Dense Z-Pinch

F. Winterberg

A dense z-pinch generated by a high voltage discharge over a corrugated helical sawtooth-shaped capillary tube with a solid core, is by shear flow stabilized against the m=0 and m=1 magnetohydrodynamic instabilities, and by rotational flow against the Rayleigh-Taylor instability. The shear- and rotational flow result from jet formation by the corrugated surface. A programmed voltage pulse can then isentropically compress the DT core to high densities, and if ignited at one end by a petawatt laser pulse, a thermonuclear detonation wave can be launched propagating along the z-pinch channel. The proposed z-pinch burn should also work without tritium as a thermonuclear detonation wave in deuterium.

Presenting Author: F. Winterberg

ThP13

Direct Conversion in a Z-Pinch IFE Reactor

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Recently, a world's record x-ray power of 0.28 PW has been measured from a Z-pinch driven imploding plasma at Sandia National Laboratory, Albuquerque, NM. This result immediately led to detailed calculations showing high fusion energy gain from a Z-pinch indirectly driven DT fusion capsule. This has fueled interest in a practical Z-pinch fusion reactor. Formidable technological obstacles must be overcome to make a practical Z-pinch driven reactor. The pulsed power system must be capable of delivering 10^7 to 10^8 pulses of energy 20-50 MJ to the load with high efficiency (> 0.1) at a rate of ~ 0.3 pulses/s. The reactor chamber must be protected from fusion reactions and byproducts. The components destroyed must be cheap enough (~$2/module).

An attractive scheme uses a MHD generator to directly convert most of the fusion energy with residual energy recovery by a steam bottoming cycle. A compact blanket absorbs most of the fusion energy and creates low temperature plasma that is used to drive a MHD generator. Rather low yields are required to convert the blanket to the ~ 1 eV plasma required to optimally drive the MHD generator. For example, a 5 GJ yield requires a flibe blanket with a thickness of only ~ 20 cm and a mass of ~ 70 kg to absorb more that 1/2 of the fusion energy.

Presenting Author: J.S. De Groot

ThP14

Motion and Stability of Liquid Metal Walls in Fusion Reactors

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Differential rotation of two touching solid shells produces only a change to the real part of the resistive wall mode frequency in a cylindrical geometry magnetic fusion device. In contrast, it is shown here that rotation of a fluid shell in contact with a solid wall tends to be destabilizing. In these computations toroidal curvature effects are neglected. A Nyquist analysis finds no additional unstable modes, but does find modified stability criteria for the resistive wall mode. Several physical effects are considered and their role in this problem evaluated. Inclusion of surface tension produces capillary waves at spatial scales much below the radius of the ARIES-RS device being modeled. Inclusion of viscosity produces a boundary layer between the inviscid flow region and the solid wall. The effects of sheared wall fluid flow are discussed.

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ThP15

Hydro*Star: A New IFE Concept Using the Fusion Chamber as a Steam Boiler

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We publicly introduce a totally new IFE power plant concept based on a self-cleaning fusion chamber having a 1 to 2-m-thick water blanket whose vaporization furnishes steam that exits directly into steam turbines.[Ref. 1] The water, operated at 100 C in a frothed-liquid state "wicked" directly from the spherical chamber wall, protects the chamber structure from irradiations from DD fusion reactions (with DT hotspots) while directly furnishing steam just like a steam boiler in a coal-fired plant. Because there are no intermediate heat exchangers, the plant thermal efficiency can be about 50%, which is significantly higher than that for nearly all other proposed systems. Higher repetition rate is also possible because the interpulse period is less restricted by vapor-condensation rates. Either solid-state or heavy-ion driver beams may be used as long as prepulse laser channeling through the minimum ambient pressure of roughly 20 ATM can prepare a suitable propagation channel. The very small annual consumption of tritium can even be purchased as a near-negligible expense, but sufficient tritium breeding can also be accomplished through internal target interactions. The result is a more flexible system that is naturally safer and that delivers energy at lower cost and at lower risk. We will discuss the scientific and engineering issues for this system, and solicit comments directed toward improvement of the concept.

[1] Charles D. Orth, "Hydro*Star: A Direct Water-Cooled DD-Fueled Inertial Fusion Reactor Concept," Internal LLNL Report ICFA 89-21 (Draft Jan. 12, 1990).

Presenting Author: C.D. Orth

ThP16

Inertial Confinement Fusion in the Neutral Plasma Imploder

S.R. Bolger

The "Neutral plasma Imploder" generates an initially cool and diffuse low temperature plasma and then electrodynamically implodes it with spherical symmetry to temperatures and pressures required for fusion. The implosion occurs in electrically neutral, field-free plasma, allowing the electrons unusual mobility to screen the ions, which reduces the required ion input energy and allows the use of high efficiency twin grid ion accelerator optics. The simple spherical reactor has no structures near the zone of reaction, scales up easily, and is easily constructed of neutron-resistant materials. The operation of the device is explained.

Presenting Author: S.R. Bolger

ThP17

Recent Studies of Star Mode IEC Devices and Possible New Directions

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Research on Inertial Electrostatic Confinement Fusion devices at the U of Illinois has focussed on use of the unique Star mode discharge [1]. This mode was selected because of the simple device geometry, requiring only one grid, combined with the excellent focusing created by the star "spokes" and the corresponding reduction in ion bombardment of the grid. Experiments have demonstrated good ion trapping due to "double" potential-well formation at the focal region with this configuration [2]. However, the extension to higher fusion reaction rates requires improved ion trapping. This in turn suggests enlarging the depth and volume of the potential trap by improved control of electron/ion currents while avoiding instabilities despite higher currents. A modification of the Star mode appears necessary in order to create the larger spread in ion angular momentum and enhanced electron flows required to achieve such enhanced trapping. Theoretical aspects of this "new" mode of operation will be discussed along with some design concepts for it's possible implementation. Initial experiments using pulsed operation to achieve the high currents needed for formation of this new mode will be described. Finally, thoughts about use of virtual electrode formation to allow removal of the focussing grid will be presented.

[1] G. H. Miley and J. Sved, "The IEC-A Plasma-Target-Based Neutron Source," *App. Radiat. Isot.*, No. 48, No. 10-12, pp. 1557-1561, 1997.

[2] Yibin Gu and George H. Miley, "Experimental Study of Potential Structure in a Spherical IEC Fusion
Device. ; *IEEE Transactions on Plasma Science*, Feb 2000.

Presenting Author: G. Miley

ThP18

Ultrafast Fusion Neutron Sources Produced From Laser Driven Explosions of Molecular Clusters

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Atomic clusters have long been studied by chemists and physicists because of the unique, intermediate position they hold between molecules and bulk solids [1]. Light induced processes in clusters can lead to photo-fragmentation and Coulombic fission, producing atom and ion fragments with a few eV of energy. However, recent studies on the photoionization of atomic clusters with high intensity ($>10^{16}$ W/cm^2), ultrafast laser pulses (~100 fs) have shown that these interactions can be far more energetic. In fact, recent results indicate that high intensity laser heating of large atomic clusters produce a super-heated microplasma which will eject ions with enormous kinetic energy when the cluster explodes [2]. This phenomenon suggests that it is possible to produce multi-keV deuterons from exploding deuterium clusters. If these ions are produced in a high density gas jet (i.e. an average ion density of $>10^{19}$ cm^{-3}) they can produce DD fusion neutrons. We have experimentally observed the production of fusion neutrons from such exploding deuterium clusters using a modest energy, high repetition rate (10 Hz) table-top laser producing 0.1 J, 35 fs laser pulses [3]. The neutrons are produced from a small volume (< 0.1 mm^3), and recent measurements of the neutron pulse duration indicate that the neutrons are emitted in a pulse lasting roughly 100 ps. If such a source of neutrons can be economically scaled to high flux per shot, it may represent a unique pulsed source of neutrons [4]. Such ultrafast sources of neutrons may enable pump-probe style experiments that examine previously unexplored time scales for materials dynamics during neutron radiation damage. In this paper, we will present recent experimental results on fusion from the exploding laser driven clusters and will examine the prospect for scaling and utilizing such a neutron source in applications of relevance to the fusion materials community.

1. A. W. Castleman, and R. G. Keese, "Gas-Phase Clusters: Spanning the States of Matter," Science 241, 36-42 (1988).
2. T. Ditmire, J. W. G. Tisch, E. Springate, M. B. Mason, N. Hay, R. A. Smith, J. P. Marangos and M. H. R. Hutchinson, "High Energy Ions Produced in Explosions of Super-Heated Atomic Clusters," Nature, 386, 54 (1997).
3. T. Ditmire, J. Zweiback, V. P. Yanovsky, T. E. Cowan, G. Hays, and K. B. Wharton, "Nuclear Fusion from Explosions of Femtosecond-Laser Heated Deuterium Cluster," Nature 398, 492 (1999).
4. L.J. Perkins, B.G. Logan, M.D. Rosen, M.D. Perry, T. Diaz de la Rubia, N.M. Ghoniem, T. Ditmire, P.T. Springer, S.C. Wilks "The Investigation of High Intensity Laser Driven Micro Neutron Sources for Fusion Materials Research at High Fluence" Nuclear Fusion, 40, 1 (2000).

Presenting Author: T. Ditmire

ThP19

Wild Cables in Fusion Plasmas (Experiment)

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The evidences for the tubular rigid-body long-living filaments (LLFs) of macroscopic size are found in tokamak and Z-pinch plasmas (few centimeters long LLFs, in a Z-pinch, and several times longer, in tokamaks). The long-livingness of similar straight filaments was proven in [1] in tracing their dynamics in a Z-pinch during almost entire discharge.

The evidences come from the results [2] of processing the visible-light images with the help of the method of multilevel dynamical contrasting (MDC) [3] (the originals are taken from experiments in the various small and moderate-size tokamaks [2] and the gaseous Z-pinch [3]; sometimes the large scale structuring may be seen even without MDC processing). The images correspond to the self-emission of the plasma (for some data from tokamak, this is the light emitted by an injected pellet and reflected by the LLFs). The reliability of the results is based on the rich statistics, considerable similarity of the LLFs observed in various regimes and various facilities, as well as the insensitivity to specific way of imaging.

The typical LLF is a straight cylindrical formation varying in length, in tokamak case, from few centimeters up to the diameter of plasma column. Such an LLF of few centimeters in diameter resembles a cable: it has a distinct inner cylinder of few millimeters diameter and an axisymmetric tubular sheath, with a distinct boundary and, often, intricate coaxial structuring. Typical dimensions of such a "wild cable" in the gaseous Z-pinch are smaller roughly by one order of magnitude. Mostly, the tubular LLFs are found at the periphery of the plasma column, however, similar structures in the central hot plasma are found as well.

The main stress is made on the identification of straight LLFs directed radially (i.e. in tokamaks, across a strong magnetic field) as this suggests the possibility of a direct (non-diffusive, non-local) energy transport toward plasma core, both in tokamaks and Z-pinches. The apparent rigidity of wild cables suggests the necessity to introduce a separate, "network" component which penetrates the conventional, "fluid" component.

The interpretation and potential implications of the present results for the fusion programs based both on conventional and innovative confinement concepts, are addressed in the accompanying paper.

[1] Kukushkin A.B., Rantsev-Kartinov V.A., Proc. 26-th EPS conf., Maastricht, Netherlands,1999,P2-087 (http://epsppd.epfl.ch/cross/p2087.htm); Proc. 3rd Symposium "Current Trends in Int. Fusion Research: Review and Assessment", Washington D.C., March 1999, Ed. E. Panarella, NRC Research Press, Ottawa, Canada (to appear).

[2] Kukushkin A.B., Rantsev-Kartinov V.A., Preprint of Kurchatov Institute, IAE 6157/6, Moscow, October 1999.

[3] Kukushkin A.B., Rantsev-Kartinov V.A., Laser and Part. Beams, 16 (1998) 445; Rev. Sci. Instrum., 70 (1999) 1387.

Presenting Author: V.A. Rantsev-Kartinov

ThP20

Wild Cables in Fusion Plasmas (Theoretical View)

A.B.Kukushkin, V.A.Rantsev-Kartinov
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A qualitative model is suggested for interpreting the observations of the tubular rigid-body long-living filaments (LLFs) of macroscopic size in tokamak and Z-pinch plasmas (see preceding abstract). The model is based on the conclusion [1] that only the quantum (molecular) long-range bonds inside LLFs may be responsible for their unbelievably high mechanical stability and survivability in a high-temperature plasma, rather than the mechanisms of a classical particles plasma. Specifically, the carbon nanotubes (CNTs), or similar structures, have been proposed to be the major microscopic building blocks of the respective microsolid component of the LLFs because CNTs are known to be produced in various low-temperature electric discharges. The well-known and just revealed properties of CNTs, and especially their assemblies, allow us to extend the survivability of the CNT-assembled macroscopic skeletons to high-temperature plasmas of high-current electric discharges.

We suggest the above survivability of the LLFs (specifically, of those of few millimeter diameter) to be caused by the channeling of the electro- magnetic (EM) energy pumped from the external electric circuit along hypothetical microsolid skeletons which are assembled during electric breakdown from individual CNTs by the originating/passing electric current. Thus, an LLF is suggested to be a "wild cable" [2] in which the skeleton works as an inner guide of the EM wave (specifically, TEM wave). The TEM wave(s) produces, by the Miller's force, a practically vacuum channel around each straight section of the skeleton and thus protect it from the access of thermal plasma particles. The model is based on the possibility of a partial conversion of the incoming EM field (in particular, almost-static poloidal magnetic field in a tokamak) into EM waves of much higher frequency. The probable mechanisms of such a conversion are discussed. The correlation of the model with the observations of strong high-frequency electric fields, both outside and inside high-temperature plasma, is found. The mechanism of a very low dissipation of the strong EM waves propagating along the CNT-assembled skeletons is discussed.

A direct (non-diffusive, non-local) energy transport toward plasma core in the wild cables opens the possibility of concentrating the energy inflow from the external electric circuit, both in magnetically and inertially confined plasmas. The status of the problem of non-local component of heat transport in tokamaks and Z-pinches is discussed.

[1] Kukushkin A.B., Rantsev-Kartinov V.A., Proc. 26-th EPS conf., Maastricht,Netherlands,1999, P2-087 (http://epsppd.epfl.ch/cross/p2087.htm); Proc. 3rd Symposium "Current Trends in Int. Fusion Research: Review and Assessment", Washington D.C., March 1999, Ed. E. Panarella, NRC Research Press, Ottawa, Canada (to appear); Proc. 17th IAEA Fusion Energy Conf., Yokohama, Japan, October 1998, IAEA-CN-69/IFP/17.

[2] Kukushkin A. B. Rantsev-Kartinov V. A. Preprint of Kurchatov Institute, IAE 6157/6, Moscow, October 1999; <arXiv.org e-Print Archive>, http://xxx.lanl.gov/abs/physisc/9910048.htm.

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ThP21

Some Applications of the Kinetic Tandem Concept*

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The Kinetic Tandem (K. T.) open-ended fusion power plant concept [1,2] employs ion or fast-neutral beams, injected into a converging solenoidal magnetic field, to create a localized region of high ion density in the vicinity of the reflection points of the injected ions. These high density regions can form the "plugs" of a tandem mirror in a manner similar to the method employed in the original tandem mirror concept [3,4]. The advantage of the K. T. over these earlier tandem-mirror concepts is that the confining magnetic field is axially symmetric, and at the same time is stable against MHD interchange modes (owing to the positive curvature of its field lines). Thus the use of complex, non-axisymmetric, fields is not required to insure MHD stability. The disadvantage of the K. T. is that to achieve a positive fusion power balance against the power required to maintain the plugs the central cell must be kilometers in length. One way to reduce the length is to employ the K. T. concept in a cusp-ended geometry. In cusp geometry MHD stability can be maintained at very high beta, thereby reducing the area of the flux bundle that must be plugged at the ring cusps, and, as a consequence, lowering the plug power requirements.

In this paper we will explore the above, and other, K T. variations, with an eye toward still further reducing the size of net-power-producing open-ended fusion systems employing the concept. One attractive possibility is to employ the K. T. plugs solely as MHD "anchors" in a conventional tandem mirror configuration, i.e. one with plug cells and thermal barriers, but now one that need only employ axially symmetric mirror fields. As shown by theory [5], a modest-pressure plasma, if it occupies an end region characterized by strong favorable field-line curvature, can stabilize a much higher pressure plasma contained in an axially symmetric mirror field, a plasma that would otherwise be unstable against MHD interchange modes. In such an embodiment the power requirements of the K. T. plugs could be substantially reduced, as compared to the case where these plugs must contain the pressure of the central-cell plasma.

*Work performed under the auspices of the U. S. Department of Energy by the Lawrence Livermore National Laboratory under contract W-7405-ENG-48.

1) R. F. Post, "Open Systems in the Quest for Fusion," Plasma Physics Reports, Vol. 23, p. 756 (Sept. 1997)
2) R. F. Post, "Mirror-Based Fusion: Some Possible New Directions," Transactions of Fusion Technology, Vol. 35, p. 40 (January 1999)
3) G. I. Dimov, V. V. Zakaidov, M. E. Kishinevskii, "Thermonuclear Confinement System with Twin Mirror Systems," Sov. J. Plasma Physics, Vol. 2, p. 326 (1976)
4) T. K. Fowler, B. G. Logan, "The Tandem Mirror Reactor," Comments Plasma Phys. Controll. Fusion, Vol. 2,, p.167 (1977)
5) D. D. Ryutov, "Axisymmetric MHD-Stable Mirrors," Physics of Mirrors, Reversed Field Pinches and Compact Tori (Proceedings of the Course and Workshop, Varenna, Italy),Vol. II, p.791 Società Italiana di Fisica (1987)

Presenting Author: R.F. Post

ThP22

Fast Wave Heating for Innovative Concepts

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Auxiliary heating of innovative concepts poses a set of problems distinct from those encountered in conventional tokamak heating. An innovative concept may have poor confinement for energetic ions, and so is poorly suited to neutral beam injection or minority ion heating by radio-frequency (RF) waves. Here the solution may be to apply a technique which results in bulk heating of electrons. However, operation at high plasma density and relatively low magnetic field also restricts the use of RF techniques such as conventional electron cyclotron heating. One RF heating technique which is well suited to high plasma density and beta is direct fast wave electron heating. In NSTX and CDX-U, electron heating at very high harmonics of the ion cyclotron frequency (HHFW heating) has been successfully demonstrated. Direct electron heating at high *frequencies* (HFFW, at 350 MHz) is also the lead candidate for RF heating in NCSX. Here we discuss fast wave heating for the ST (HHFW), the extension to devices with higher confining magnetic field such as NCSX (HFFW), and possible applications to other concepts such as the FRC.

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